

Research Reactors

Volume 1



SELECTED REFERENCE MATERIAL
UNITED STATES ATOMIC ENERGY PROGRAM

1955

Research Reactors



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Research Reactors



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**SELECTED REFERENCE MATERIAL
ON ATOMIC ENERGY**

- VOLUME ONE Research Reactors**
- VOLUME TWO Reactor Handbook: Physics**
- VOLUME THREE Reactor Handbook: Engineering**
- VOLUME FOUR Reactor Handbook: Materials**
- VOLUME FIVE Neutron Cross Sections**
- VOLUME SIX Chemical Processing and Equipment**
- VOLUME SEVEN Eight-year Isotope Summary**
- VOLUME EIGHT Information Sources**

VOLUME ONE

Research Reactors

Light Water Moderated

TYPE I *Homogeneous—Enriched Fuel*

TYPE II *Heterogeneous—Enriched Fuel*

TYPE III *Heterogeneous—Enriched Fuel*

Light Water and Oil Moderated

Heterogeneous—Enriched Fuel

Heavy Water Moderated

Heterogeneous—Enriched Fuel

Graphite Moderated

Heterogeneous—Natural Fuel

UNITED STATES OF AMERICA

GENEVA: AUGUST 1955

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Foreword

Interchange of scientific and technical knowledge will greatly facilitate the work of the scientists and engineers whose skills will be devoted to the future development of the peaceful uses of atomic energy.

The United States has made available to the world's scientific community a large body of such data. In honor of this historic Conference and to stimulate further exploration and development of the beneficial applications of nuclear energy, the United States Atomic Energy Commission has prepared this special collection of technical data for the use of the delegates and the nations represented.

The purpose of this collection is to provide information concerning the ways that we have found in which fissionable materials can be put to work in nuclear reactors for research purposes and for the production of power and radioisotopes.

It is our sincere hope that this material will be of practical value to the men and women of science and engineering in whose hands the great power of the atom is becoming a benign force for world peace.

Levin L. Strauss

Chairman, U.S. Atomic Energy Commission

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Introduction

Within the brief span of 10 years nuclear reactors have proved themselves an indispensable research tool in a variety of fields. Up until 1953, all reactors in the United States were owned by the government and were operating on government sites. Within recent years, however, a considerable number of private research institutions and universities have planned, built, and operated research reactors on their own sites with active encouragement from the Atomic Energy Commission (AEC). In addition there are upwards of a dozen industrial firms in this country which are planning, building, or actually operating various types of nuclear reactors. A valuable backlog of reactor design and operating experience has accumulated and is available to assist groups to enter the field. On the basis of this experience, some general considerations on the uses, classifications, performance, costs, and safety problems of nuclear reactors are presented on the pages that follow.

Uses of Reactors. The practical applications of nuclear research reactors may be divided into the following categories: (1) reactor technology, (2) physics research, and (3) biomedical and industrial research.

Reactor technology. Reactors are an exceedingly useful source of neutrons for a variety of experiments in the study of reactor physics and of the properties of neutrons themselves. Reactors also provide an excellent means for studying radiation damage, giving personnel experience with radiation and the methods used for detecting and counting neutrons or gamma rays. Among the many possible reactor experiments are the following:

1. Critical-multiplication-approach studies.
2. Control-rod calibrations.
3. Critical behavior studies (reactor period as a function of excess reactivity).
4. Temperature-coefficient measurements and studies of the effect of temperature on reactivity.
5. Studies of neutron-flux distribution.
6. Studies of the effects of neutrons on materials (radiation damage).
7. Shielding studies.

Physics research. Reactors are excellent tools for performing fundamental experiments in the field of physics. Studies of the fission process and measurements of neutron cross sections are among the commonly performed experiments possible with a research reactor. The following are typical examples of experiments designed to study the fission process:

1. Determination of neutron-energy distribution from the fission of U^{235} .
2. Observation of the number and lifetimes of neutrons and the intensity of gamma rays occurring after fission.
3. Determination of the total energy of fission by calorimetric methods.
4. Measurement of the velocity distribution of the fission fragments.

Examples of neutron cross-section measurements include:

1. Fission, absorption, and scattering cross sections at various neutron energy levels.
2. Total cross section for a large number of elements by using a neutron converter in the beam hole as a fast-neutron source and a proton-recoil chamber as a neutron spectrometer.

Biomedical and industrial research. Neutron flux from reactors provides the basis for many

interesting biological and industrial radiation-damage studies. Furthermore, radioactive isotopes, obtained by irradiation of samples with neutrons in a reactor, are widely used for medical, biological, and industrial research with outstanding results. Small amounts of the solution in a homogeneous reactor itself furnish excellent sources of rare isotopes. Among examples of the use of reactor neutron flux and radioisotopes for biomedical and industrial research are the following:

1. Experiments to determine the biological and physical effects of thermal neutrons and hard gamma rays.
2. The use of radioactive tracers to follow accurately the progress of foods and minerals in plants and animals.
3. Studies of fundamental blood processes, gland functions, and animal metabolism by means of radioisotopes.
4. Radiation-damage studies of various materials by irradiation in a reactor.
5. Investigation of self-diffusion of atoms through a crystal lattice.
6. Irradiation of silicon and germanium crystals to investigate their electrical properties.
7. Sterilization of food products by means of spent reactor fuel elements, reactor fission products, or radioisotopes.

Reactor Classification. Nuclear reactors may be classified in several different ways, depending on the type and arrangement of fuel, moderator, and coolant used and on the speed of the neutrons sustaining the fission reaction. Materials used to reduce the neutron energy, known as *moderators*, are primarily graphite, light water, and heavy water. Active fuel materials may be *natural* uranium; *enriched* uranium, in which the U^{235} fraction has been increased; plutonium (Pu^{239}); or uranium (U^{233}) resulting from fission of thorium (Th^{232}). The last two are artificial, reactor-produced elements.

The arrangement of moderator and fuel provides a basis of classification. In a *heterogeneous*, or solid-fuel, reactor the fuel is fixed in a regular pattern (lattice) within the moderator. In a *homogeneous* reactor the fuel and moderator are intimately mixed in the form of a solution, whether aqueous or liquid salts or metals.

Reactors operated above minute power levels must use some form of cooling to remove the heat produced. Light-water, heavy-water, or air coolants are generally used; either natural convection or forced circulation is employed.

Reactors are further classified according to the speed or energy of the neutrons that cause fission. Neutrons of about 0.025 ev are called *slow* or *thermal*; neutrons from about 1 to 1000 ev are known as *intermediate*; and neutrons with energies greater than about 1000 ev are called *fast*. A fast reactor does not use a moderator to slow down the fission neutrons.

Finally, reactors may be classified according to their purpose. *Research* reactors are built primarily to supply neutrons for physical research and radioisotope manufacture. *Production* reactors are built to manufacture fissionable materials by conversion of nonfissionable (fertile) materials. Reactors capable of providing useful power outputs are known as *power* reactors.

The present volume is confined to a description of six classes of research reactors operating in the United States. These are presented in six chapters, in accordance with the following classification:

1. Light-water-moderated reactors.
 - Type I: Homogeneous—enriched fuel (water-boiler type). Chap. 1.
 - Type II: Heterogeneous—enriched fuel (pool type). Chap. 2.
 - Type III: Heterogeneous—enriched fuel [materials-testing reactor (MTR)]. Chap. 3.
2. Light-water- and oil-moderated reactors.
 - Heterogeneous—enriched fuel [nuclear-testing reactor (NTR)]. Chap. 4.
3. Heavy-water-moderated reactors.
 - Heterogeneous—enriched fuel [Argonne research reactor (CP-5)]. Chap. 5.
4. Graphite-moderated reactors.
 - Heterogeneous—natural fuel [Brookhaven National Laboratory (BNL)]. Chap. 6.

Performance and Costs. It is of interest to compare the performance and costs of natural-uranium research reactors with reactors using enriched fuels, since the United States has had considerable experience with both types. Some characteristics of natural-uranium research reactors

are shown in Table I-1, and equivalent data for enriched-uranium reactors are given in Table I-2.

A comparison of the performance of various reactors measured in terms of specific power and thermal-neutron flux is shown in Table I-3. The approximate cost of the reactors, not including

reaction will get out of control, that the power will rise to a very high level in an extremely short time, and that parts of the reactor will melt or even vaporize. Based on past experience, there is very little chance of any existing or proposed research reactor destroying itself as a result of such

Table I-1 Research Reactors Fueled with Natural Uranium

Designation	Date	Fuel	Moderator	Power level
CP-1	1942	6 tons U metal	385 tons C	100 watts
CP-2	1943	40 tons U oxide 10 tons U metal	472 tons C	2 kw
X-10	1943	42 tons U oxide		
CP-3	1944	35 tons U metal	620 tons C	3,800 kw
NRX	1947	3 tons U metal	6.5 tons D ₂ O	300 kw
BNL	1950	10 tons U metal	17 tons D ₂ O	30,000 kw
		100 tons U metal	730 tons C	30,000 kw

Table I-2 Research Reactors Fueled with Enriched Uranium

Designation	Date	Amount enriched U ²³⁵ in fuel	Moderator	Power level
Heavy-water type:				
CP-3'	1950	4 kg U + Al	7 tons D ₂ O	300 kw
CP-5	1953	1.3 kg U + Al	6.5 tons D ₂ O	1000 kw
Water boilers:				
LOPO	1944	0.5 kg UO ₂ SO ₄	H ₂ O	0.05 watt
HYPO	1944	0.8 kg UO ₂ SO ₄	H ₂ O	6 kw
SUPO	1951	0.8 kg UO ₂ SO ₄	H ₂ O	35 kw
North Carolina	1953	1 kg UO ₂ SO ₄	H ₂ O	10 kw
MTR type:				
LITR	1950	3 kg U + Al	H ₂ O	2000 kw
BSF	1951	3.6 kg U + Al	H ₂ O	100 kw
MTR	1952	4 kg U + Al	H ₂ O	30,000 kw
NTR	1951	2.5 kg U + Al	H ₂ O + oil	30 kw

fuel costs, is also listed for comparison. The cost figures in general are for the first model of each reactor and hence include some reactor- and site-development costs. The same reactors could in many cases be built at the present time at a considerably lower figure.

Safety Problems. The major hazard affecting reactor safety is the possibility that the nuclear

a runaway nuclear reaction. The water systems of enriched-fuel light-water-moderated reactors are generally designed to have a strong negative temperature coefficient, so that they tend to shut down automatically following a sudden increase in reactivity. This has been demonstrated experimentally in a number of reactors of this type. In addition, the designer of a particular reactor makes adequate provisions for automatic

Table I-3 Performance and Cost of Research Reactors

Reactor	Date	Specific power, kw/ton of fuel	Avg thermal-neutron flux	Cost, millions
CP-1	1942	0.002	4×10^6	\$ 1.5
CP-2	1943	0.04	1×10^6	2.0
X-10	1943	110	5×10^{11}	5.2
CP-3	1944	100	5×10^{11}	2
HYPO	1944	7,000	1×10^{11}	0.5
NRX	1947	3,000	2×10^{12}	10
CP-3'	1950	68,000	2×10^{12}	2
BNL	1950	300	5×10^{12}	25.5
LITR	1950	500,000	1×10^{13}	1
SUPO	1951	50,000	1×10^{12}	0.5
BSF	1950	37,000	1×10^{12}	0.25
MTR	1952	7,000,000	2×10^{14}	17
NTR	1951	11,000	9×10^{11}	~0.4
NCSR	1953	9,100	2×10^{11}	0.5
CP-5	1954	910,000	2×10^{13}	3

shutdown in the event of any number of minor and major emergencies.

In the event of a reactivity surge, the possibility exists that ruptured fuel elements might release radioactive materials. However, when properly designed and operated, research reactors are quite stable and controllable from a nuclear point of view.

Reactor Abbreviations. A number of reactor abbreviations have been used freely in this introduction and throughout the remainder of the volume. These abbreviations and their equivalents are listed here for reference.

ANL: Argonne National Laboratory.

Aquarium: Pool-type reactor at Geneva (light-water-moderated, heterogeneous—enriched).

BNL: Brookhaven National Laboratory reactor (graphite-moderated, heterogeneous—natural).

BSF: Bulk-shielding facility
BSR: Bulk-shielding reactor

CP-1, CP-2: Chicago Pile (graphite-moderated, heterogeneous—natural).

CP-3: ANL heavy-water reactor (heavy-water-moderated, heterogeneous—natural).

CP-3', CP-5: Argonne research reactor (heavy-water-moderated, heterogeneous—enriched).

HYPO: High-power water-boiler reactor at Los Alamos (light-water-moderated, homogeneous—enriched).

LITR: Low-intensity test reactor at Oak Ridge (light-water-moderated, heterogeneous—enriched).

MTR: Materials-testing reactor at Arco, Idaho (light-water-moderated, heterogeneous—enriched).

NAA: North American Aviation, Inc., site of reactors.

NCSR: North Carolina State College at Raleigh, site of a water-boiler reactor.

NRX: Chalk River, Canada, site of heavy-water-moderated natural-uranium reactor.

NTR: Nuclear test reactor at Knolls Atomic Power Laboratory (light-water- and oil-moderated, heterogeneous—enriched).

ORNL: Oak Ridge National Laboratory.

Pool reactor: A light-water-moderated heterogeneous—enriched-fuel reactor immersed in a pool of water.

SUPO: Super-power water-boiler reactor at Los Alamos (light-water-moderated, homogeneous—enriched).

Water boiler: A light-water-moderated homogeneous—enriched-fuel reactor.

X-10: Reactor at Oak Ridge National Laboratory (graphite-moderated, heterogeneous—natural).

pool reactor at Oak Ridge (light-water-moderated, heterogeneous—enriched).

CHAPTER 1

Light-water-moderated Reactor

TYPE I *Homogeneous—Enriched Fuel*

COMPILED BY NORTH AMERICAN AVIATION, INC.

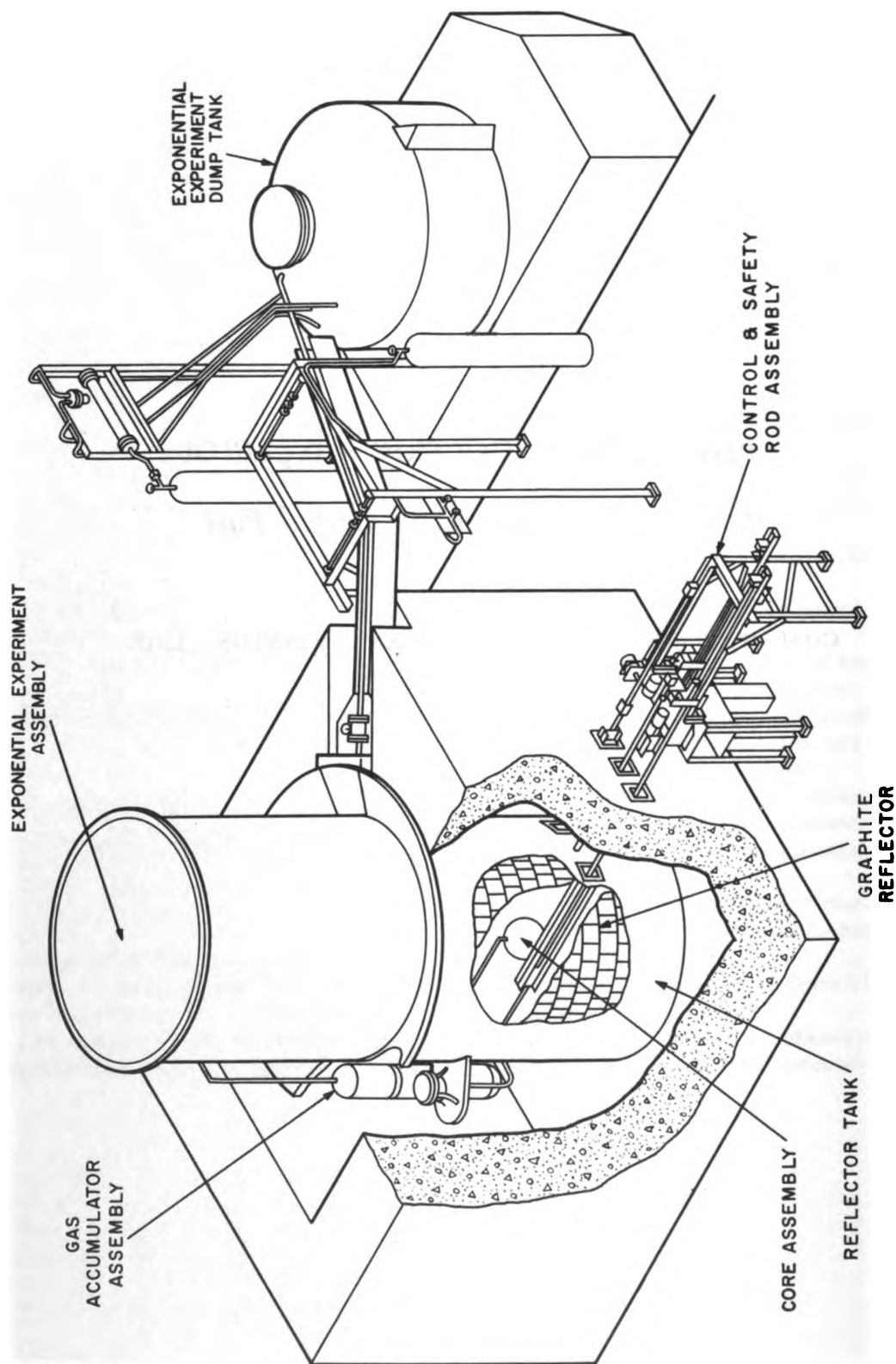


Fig. 1-1 One-watt solution-type reactor, isometric.

CHAPTER 1

Light-water-moderated Reactor

TYPE I *Homogeneous—Enriched Fuel*

A self-sustaining neutron fission chain reaction cannot be maintained in the natural uranium-water system. When uranium enriched in the U^{235} isotope began to be available from the Oak Ridge separation plants early in 1944, however, it became possible to consider realistically the construction of enriched uranium-water reactors. The first reactor of this type, the LOPO Unit, was built in Los Alamos in 1944. It operated at a power of about 0.05 watt and used 14 per cent U^{235} . It was later converted to operate at successively higher powers: the HYPO Unit at 6 kw, and the SUPO Unit at 45 kw.

The nuclear-research reactors described in this chapter use uranium enriched in the U^{235} isotope as fuel. The uranium is present as uranyl nitrate or uranyl sulfate in solution in ordinary (light) water. Because the uranium is evenly dispersed in the water solution, these reactors are called homogeneous solution-type reactors. These reactors are also often called *water boilers* because during operation there is a slow decomposition of the water and an evolution of hydrogen and oxygen which causes a bubbling similar to that of boiling. Boiling does not occur, however, and the solution temperatures are normally kept below about 80°C by circulating a water coolant through coils in the sphere. Operation is normally at or just below atmospheric pressure.

Solution-type reactors are characterized by the small size of the reactor core—typically stainless-steel spheres about 1 ft in diameter containing about 14 liters of the fuel solution. The water in which the uranium is dissolved serves as both

solvent and moderator. A layer of graphite or beryllium oxide about 2 ft thick serves as a reflector around the core. From 2 to 5 ft of concrete, depending upon the power level, serves as a shield around the reflector. Various experimental facilities—thermal columns, beam tubes, isotope-production tubes, etc.—are arranged about the core with appropriate openings through the shield and reflector according to the purpose of the reactor.

The reactor is controlled by rods containing cadmium or boron, which pass either through the core or immediately adjacent to it. The neutron-level information necessary for reactor control is provided by ion and fission chambers.

A unique feature of solution-type reactors is the decomposition of the water in the core by the action of the highly energetic fission fragments. This phenomenon makes desirable, at least at higher power levels, a recombination system for the hydrogen and oxygen and a means for returning the recombined water to the core. The radioactive fission gases, xenon and krypton, are also evolved from the solution, and therefore it is necessary to place the recombination system in a shielded area. This gas system must be either completely enclosed or carefully vented. These radioactive fission gases provide, however, a neutron-free source of gamma radiation and can be used for experiments where such radiation is desired. It is also necessary to keep the hydrogen and oxygen concentration below the explosive limit by adequate circulation of the gases through the gas system.

Although the reactors described here operate on highly enriched uranium (90 per cent U^{235}), they can be satisfactorily operated with uranium of lower enrichments. The minimum enrichment that can be used with the reactor designs and core sizes described in this report is in effect limited by the solubility of uranium sulfate (or nitrate) in water. The maximum practical concentration of uranium sulfate is 550 g of uranium per liter of water at 15°C. About 1200 g of U^{235} is required for a 50-kw solution-type reactor at enrichments between 15 and 20 per cent. Since there are 14 liters of solution in the core, it is possible to dis-

for the Armour Research Foundation, Chicago, Illinois, which is to be capable of operation at 50 kw, is described briefly.

A precise statement of the cost of a nuclear reactor must include consideration of building costs, because the reactor is usually housed in a specially designed building. Rather arbitrary decisions must be made as to what portion of the costs should be assigned to the reactor and what portion to the building. Reactor foundations must be provided as well as space for the control console, coolant pumps, heat exchangers, waste disposal, etc.

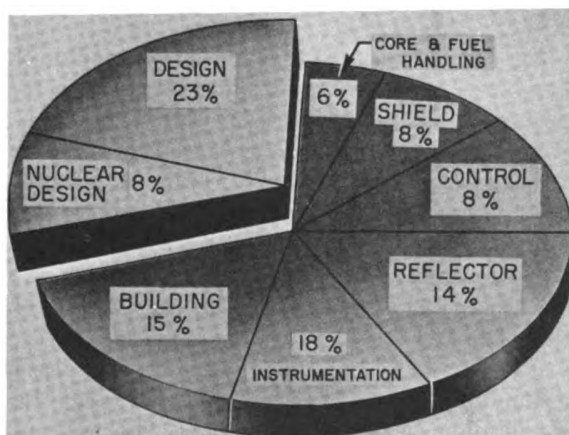


Fig. 1-2a Cost breakdown.

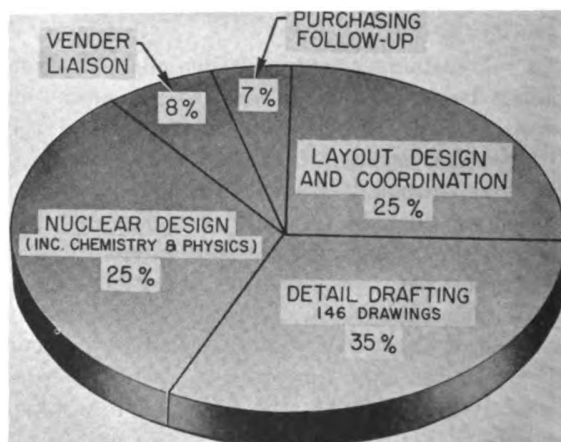


Fig. 1-2b Design breakdown.

solve $14 \times 550 = 7725$ g of uranium. A minimum enrichment of $1200/7725$, or 15.5 per cent, is therefore possible. By increasing the size of the core sphere it would be possible to use even lower enrichments, although below 10 per cent enrichment the nuclear properties of the uranium-water solution begin to approach a limit where the reactor can no longer be made critical.

Another reactor of this type has been built at North Carolina State College for operation at 10 kw.

The two reactors described in detail in this chapter are the water-boiler neutron source (WBNS), which operates at 1 watt, and the Livermore research reactor (L-3), which operates at 500 watts. The WBNS is in Downey, California, and the L-3 at Livermore, California. The design of a third reactor of this type, the L-8

For example, the cost, in 1952, of the 1-watt WBNS was \$75,000, with an additional cost of \$10,000 for a simple shed-type structure 20 by 40 by 24 ft high (with a 2-ton traveling crane). A relative breakdown of the component costs of this reactor is shown in Fig. 1-2a and a breakdown of design effort in Fig. 1-2b.

A 50-kw solution-type reactor which is presently being installed at the Armour Research Foundation in Chicago will cost about 40 per cent of the total \$500,000 cost of the reactor facility. The \$200,000 reactor cost includes \$25,000 for the dense concrete shield but does not include the preparation of the foundation beneath the reactor. The \$300,000 for the rest of the facility includes a reactor room 48 by 72 ft by 30 ft high, plus a laboratory and office area of about 10,000 sq ft.

ONE-WATT SOLUTION-TYPE REACTOR

General Design Features. The 1-watt solution-type reactor consists of a core of enriched uranyl nitrate in a 1-ft-diameter stainless-steel sphere encased in a cylinder of reactor-grade

safety system includes two control rods and two gravity-powered safety rods. The control rods are positioned according to indications and recordings on the control-panel instruments, which are fed, in turn, by signals from the neutron-detecting device located around the reactor tank.

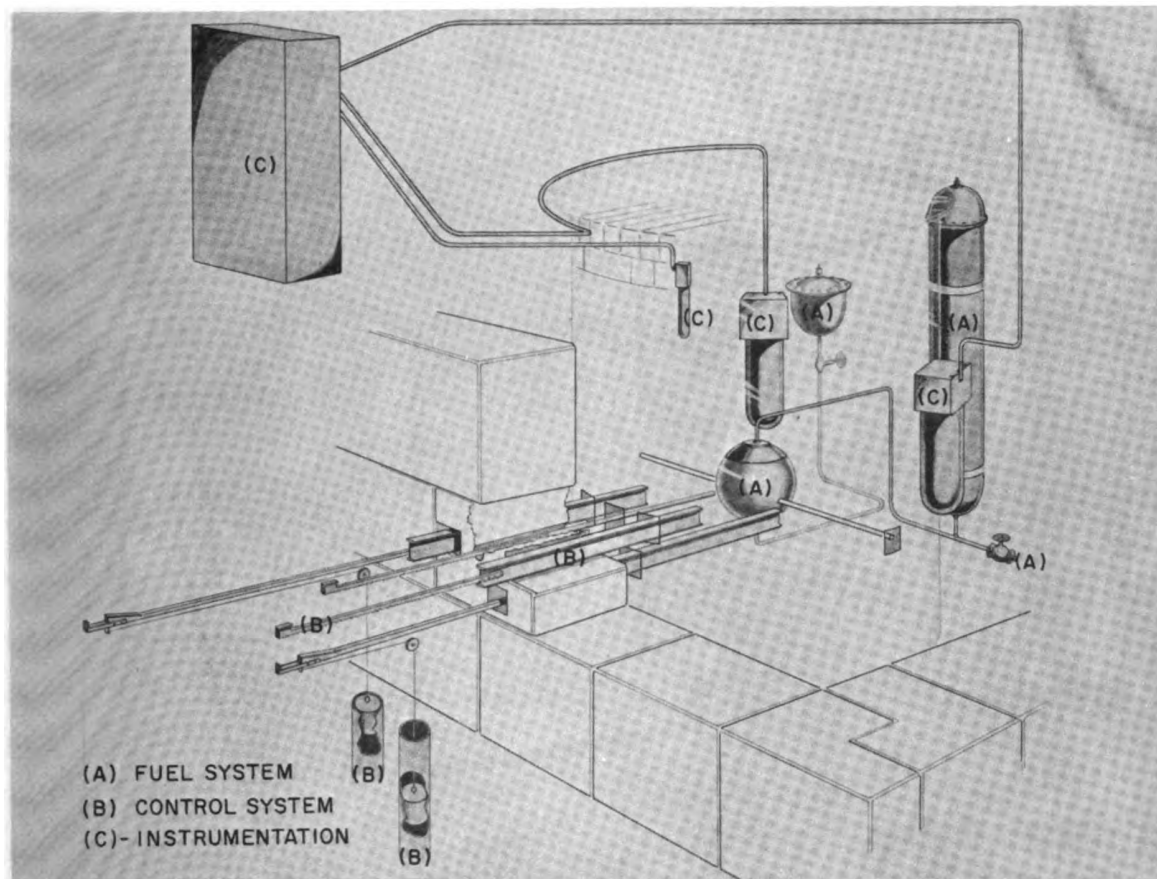


Fig. 1-3 Solution-type reactor systems.

graphite 5 ft in diameter by 6 ft high. The graphite cylinder serves as a reflector and vertical thermal column. The entire cylinder is surrounded by a concrete-block radiation shield 2 ft thick. This reactor is designated the water-boiler neutron source (WBNS).

Figure 1-3 shows the major systems of the reactor: (A) fuel and gas disposal, (B) control and safety, and (C) instrumentation. The fuel and gas-disposal system consists of the core tank, the fuel-mixing tank, the gas-accumulator tank, and valves and connecting piping. The control and

When operated at a power level of 1 watt, the WBNS supplies a maximum thermal flux of approximately 4×10^7 neutrons/cm²/sec at the center of a test hole through the spherical core. This type of reactor possesses strong inherent safety features resulting from a large negative temperature coefficient and a negative power coefficient of reactivity. These negative coefficients are sufficient to shut down the reactor in the event of an accidental release of excessive reactivity. The shutdown occurs with a relatively small release of energy.

Core and Fuel-handling System. The core and fuel-handling system, consisting of core tank, mixing tank, and accumulator tank, is shown in Fig. 1-4.

The core tank is a 12-in.-diameter sphere constructed from two spun hemispheres of type 347

connected to the core tank by tubing through a plug gate valve.

During operation of the reactor, the gases evolved from the fission process and from the decomposition of the solution are collected in the gas-accumulator tank. The accumulator tank is

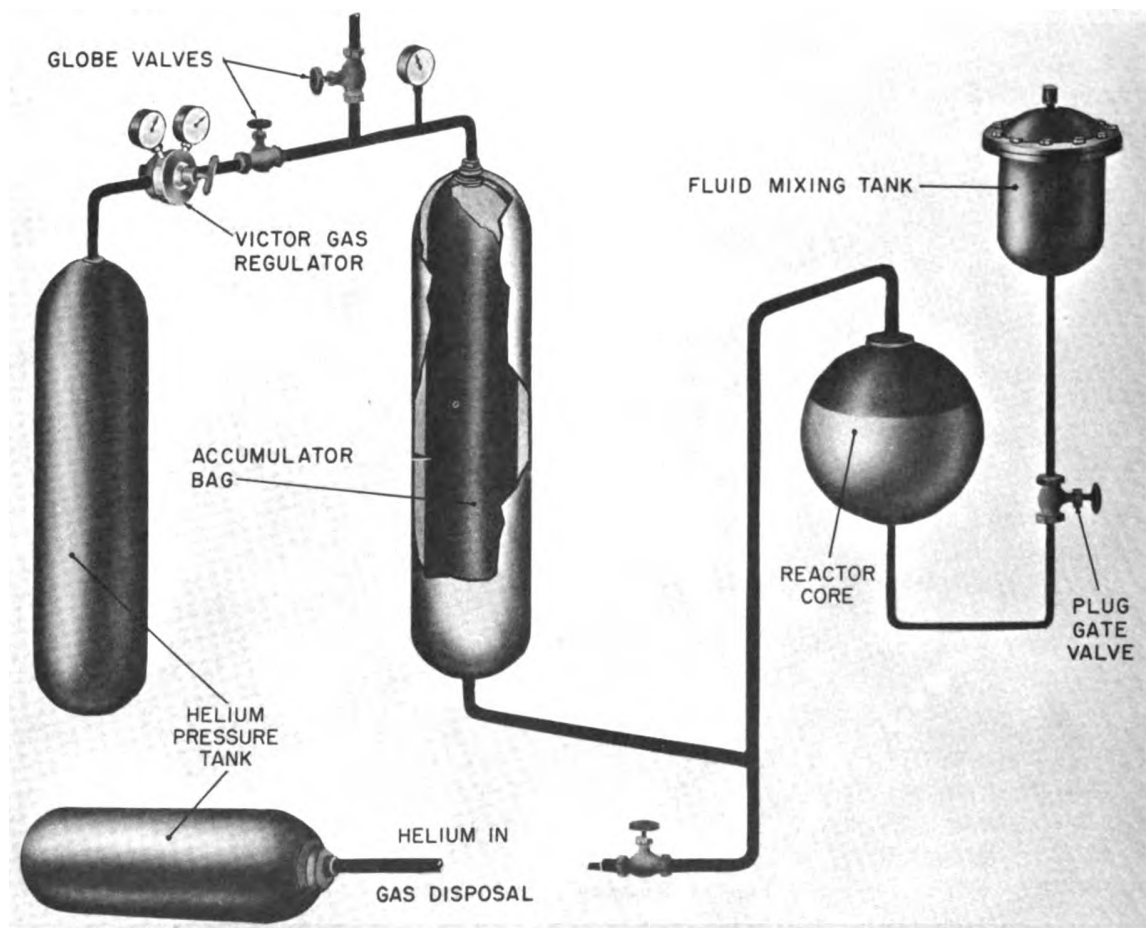


Fig. 1-4 Fuel-handling system.

stainless-steel sheet $\frac{1}{16}$ in. thick. The tank has a volume of 14.38 liters. A central exposure facility in the core has been formed by inserting a tube $1\frac{1}{8}$ -in. ID through the sphere with its center line 3 in. below the horizontal diameter of the sphere.

The mixing tank is made of 347 stainless steel highly polished for cleanliness and is fitted with a sealed cover which allows the tank to be pressurized or evacuated. The mixing tank is con-

made of stainless steel and has a volume of approximately 40 liters. Inside the tank is a neoprene-rubber bag which, when expanded, completely fills the tank, expelling all its contents as it expands. The bag is connected through the top of the accumulator tank to a helium pressure tank. When sufficient fission gases have been collected between the outside of the bag and the inside of the accumulator tank, the bag is expanded and the fission gases are forced into a dis-

posal tank which can be removed from the system.

Graphite Reflector. Graphite for the reflector was obtained in rough extrusions about $4\frac{1}{2}$ in. high, 12 in. wide, and 4 ft long. The extrusions were machined on four sides and one end to accurate, uniform dimensions (Fig. 1-5). The

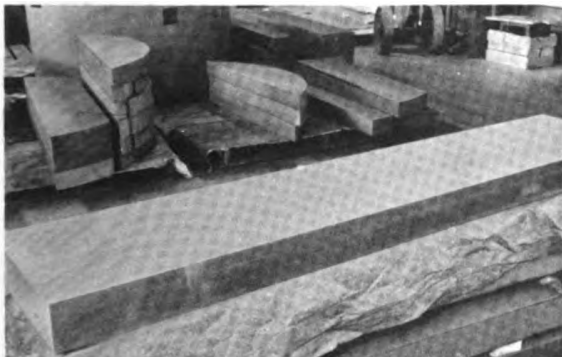


Fig. 1-5 Extruded reflector graphite.

blocks are selectively assembled so that the layers of graphite are composed of blocks having the same thickness. By this method, uniform load distribution is assured. Special care was given to shaping the blocks which are adjacent to the sphere to leave a minimum of air space. Each layer of blocks is laid with joints in adjacent layers 90° apart. The principal experimental facility for this reactor is a tank placed directly on top of the graphite. The top surface therefore was carefully leveled and kept scrupulously clean during final assembly. The steel tank containing the graphite (Fig. 1-6) has walls $\frac{1}{4}$ in. thick and is supported by 6-in. channel beams welded to the bottom. Openings are provided on the sides to match the locations of control rods and special stringers in the reflector graphite. A strong flange is provided at the top to anchor the experiment tank against movement due to horizontal forces. The reflector tank is mounted on a structural-steel base which is furnished with lifting lugs capable of handling the total load of graphite and tank without damage. Research facilities (Fig. 1-7) are provided by removable stringers of graphite and an access hole through the reflector to the exposure hole, permitting maximum irradiation of experimental pieces at the center of the core. The

removable stringers are shielded by special removable close-tolerance plugs made of wood tipped with lead and cadmium.

Safety and Control Rods. Reactivity control is maintained by means of two safety rods, a coarse-control rod, and a fine-control rod located as shown in Fig. 1-8. These rods move horizontally through the concrete shield and into the graphite reflector adjacent to the stainless-steel sphere.

Each safety rod is constructed of two $\frac{1}{4}$ -in.-thick strips of boral attached to aluminum channel beams. This forms an I beam about 3 ft long by 4 in. high.

The safety rods are held in the OUT position by a trigger mechanism, which can be tripped either manually or by electrical activation from the control panel. Metal counterweights acting under the force of gravity pull the rods to the IN position to shut down the reactor. When the rods

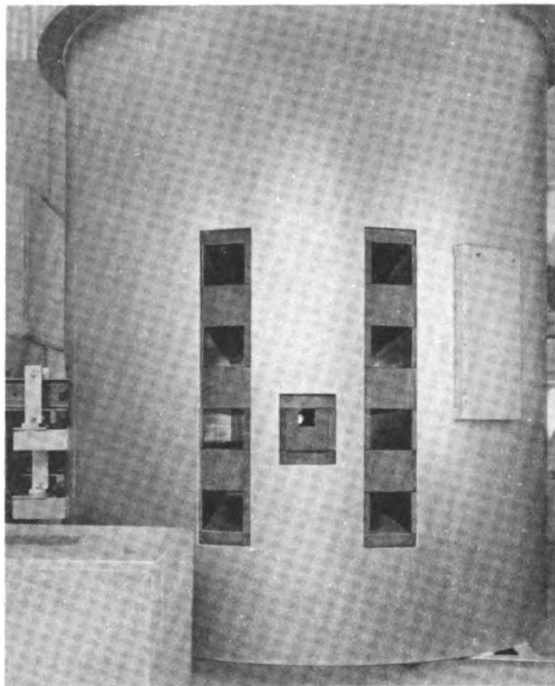


Fig. 1-6 Reflector tank.

are within 6 in. of their total travel, the counterweights are arrested and wedges on the control rods contact spring-backed brake shoes to absorb the momentum of the rods and stop them before they contact the graphite in the reflector.

The coarse-control rod is similar to the safety rods, but with cadmium sheet instead of boral as the neutron absorber. This rod is driven by a reversible electric motor. The traverse of the rod is 80 cm, and approximately 155 sec is required

electrometers monitoring the neutron level of the reactor and a variable standard signal. The output of the amplifier drives a two-phase motor, which in turn drives the fine-control rod through a suitable gear train. Each of the control rods is

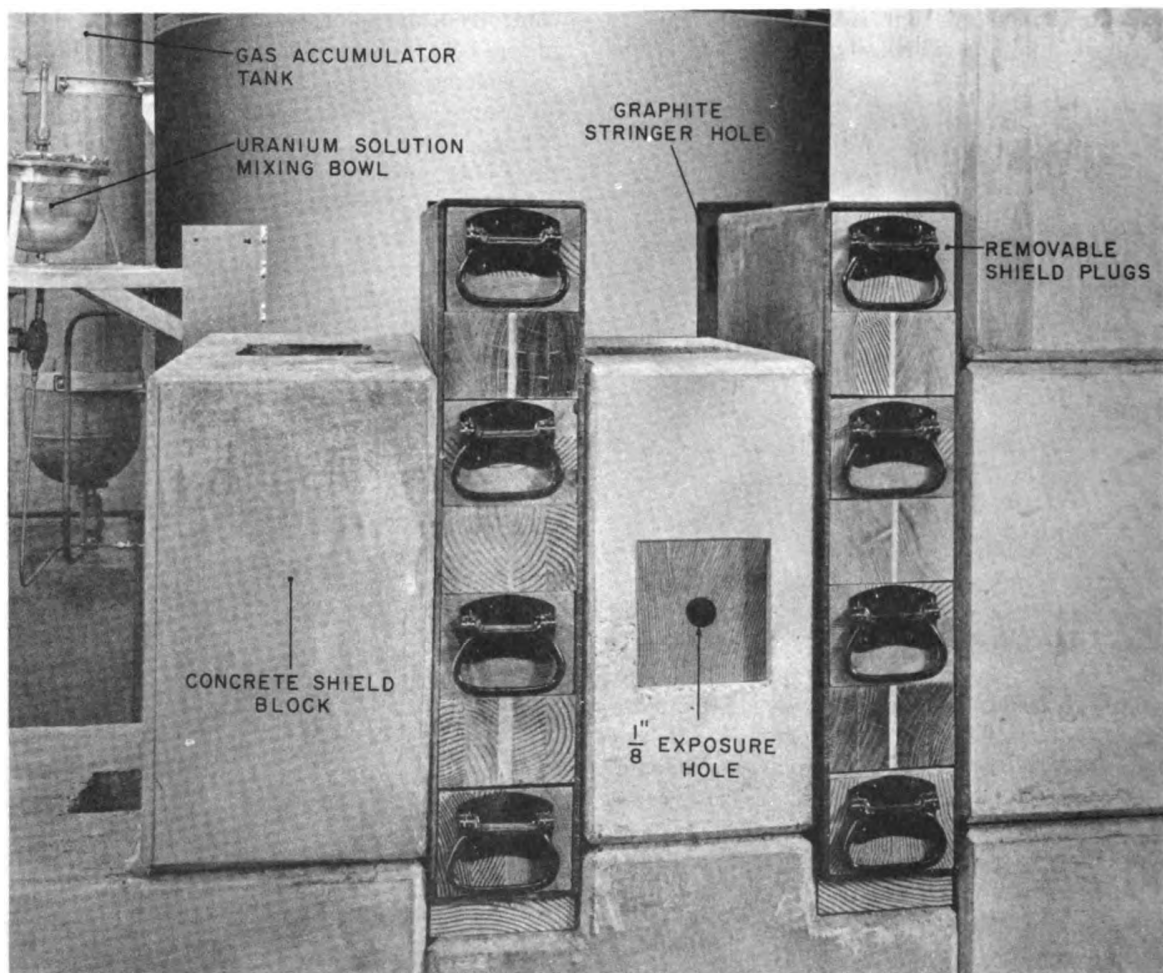


Fig. 1-7 Access facilities.

for the movement from the IN position to the OUT position.

The fine-control rod, which is used as an automatic power-regulating rod, consists of a 1-in.-diameter steel pipe with provision at the end for insertion of varying amounts of cadmium. This rod controls approximately 0.1 per cent in reactivity. For automatic control, a servoamplifier is utilized. The input to the servoamplifier is the difference between the signal from one of the

provided with a selsyn remote indicator located on the control panel.

Instrumentation. Figure 1-9 shows a view of the instrument-control panel, located in a laboratory area immediately adjacent to the reactor. A block diagram of the instrumentation for control of the reactor is shown in Fig. 1-10. Four neutron-flux measuring instruments, placed at different positions around the graphite reflector, are

used for monitoring the power level of the reactor. Two of these are BF_3 ionization chambers connected to vibrating-reed electrometers. The others are a boron-lined counter connected to a counting-rate meter and a BF_3 proportional counter connected to a scaler. A recorder

for the reactor. Concrete blocks 2 ft square by 1 ft 10 in. high were selected. Vertical joints were staggered at assembly, and the corner blocks were interlocked. Some of the blocks were designed in special shapes to give passage for control rods or the movable graphite stringers.

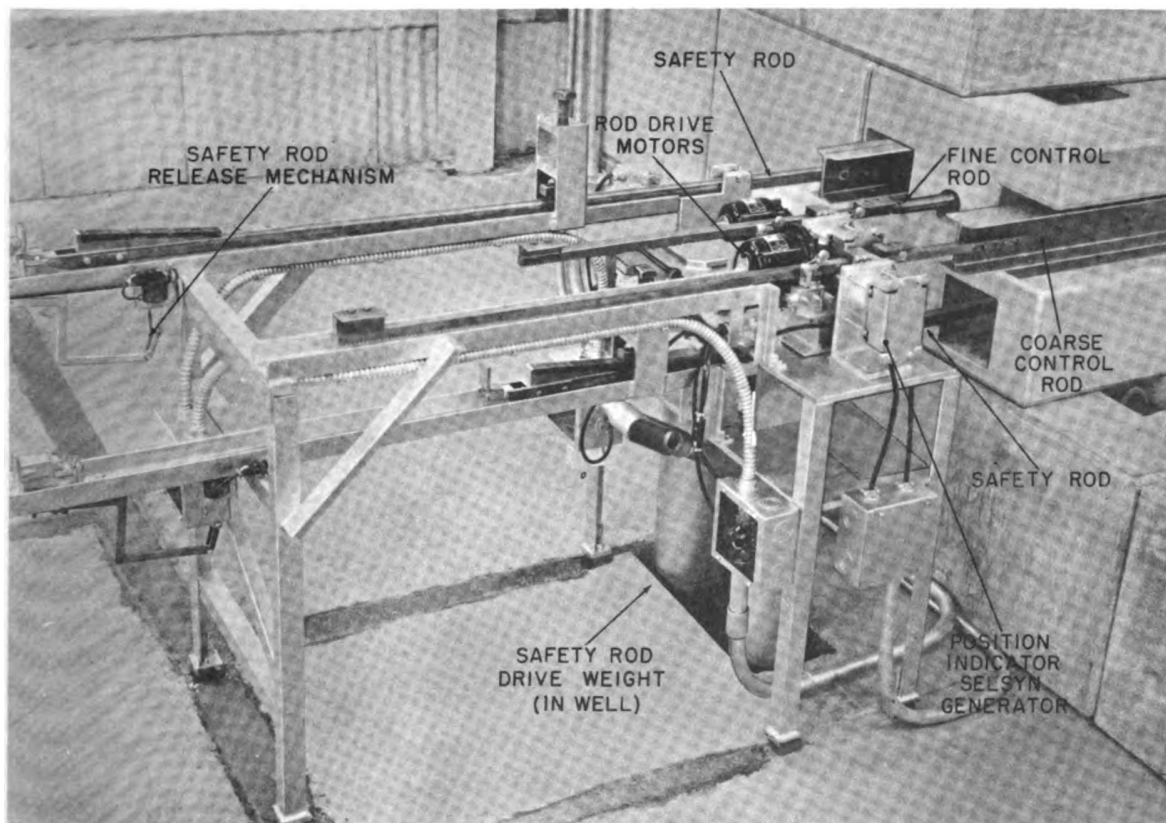


Fig. 1-8 Control- and safety-rod mechanism.

located on the control panel continuously records the signal from one of the ionization chambers to provide a permanent record of the reactor power level. Sensitive relays, of a moving-contact type (Sensitrol), are connected to the output of each electrometer and to the counting-rate meter. Operation of any one of these three relays will release the safety-rod latches and shut down the reactor.

Shield. Figure 1-11 shows the biological shield which was designed to be adequate for the radiation field present, yet demountable in case varying experimental arrangements of shield should be required or in case of a complete change of site

During assembly, the key blocks which were adjacent to the control and safety rods and the special blocks opposite the experiment holes in the reflector were first located and the remaining blocks filled in to suit. It was necessary to level and shim all of the blocks of the lowest layer until the top surfaces of all were in one plane before progressing to the next layers. By doing this a successful assembly was achieved, and a source rod could then be passed through shield block and reflector and into the exposure hole in the sphere; the control rods passed freely within graphite walls and the removable stringers of graphite could be transferred in and out of the reflector without difficulty from outside the shield.

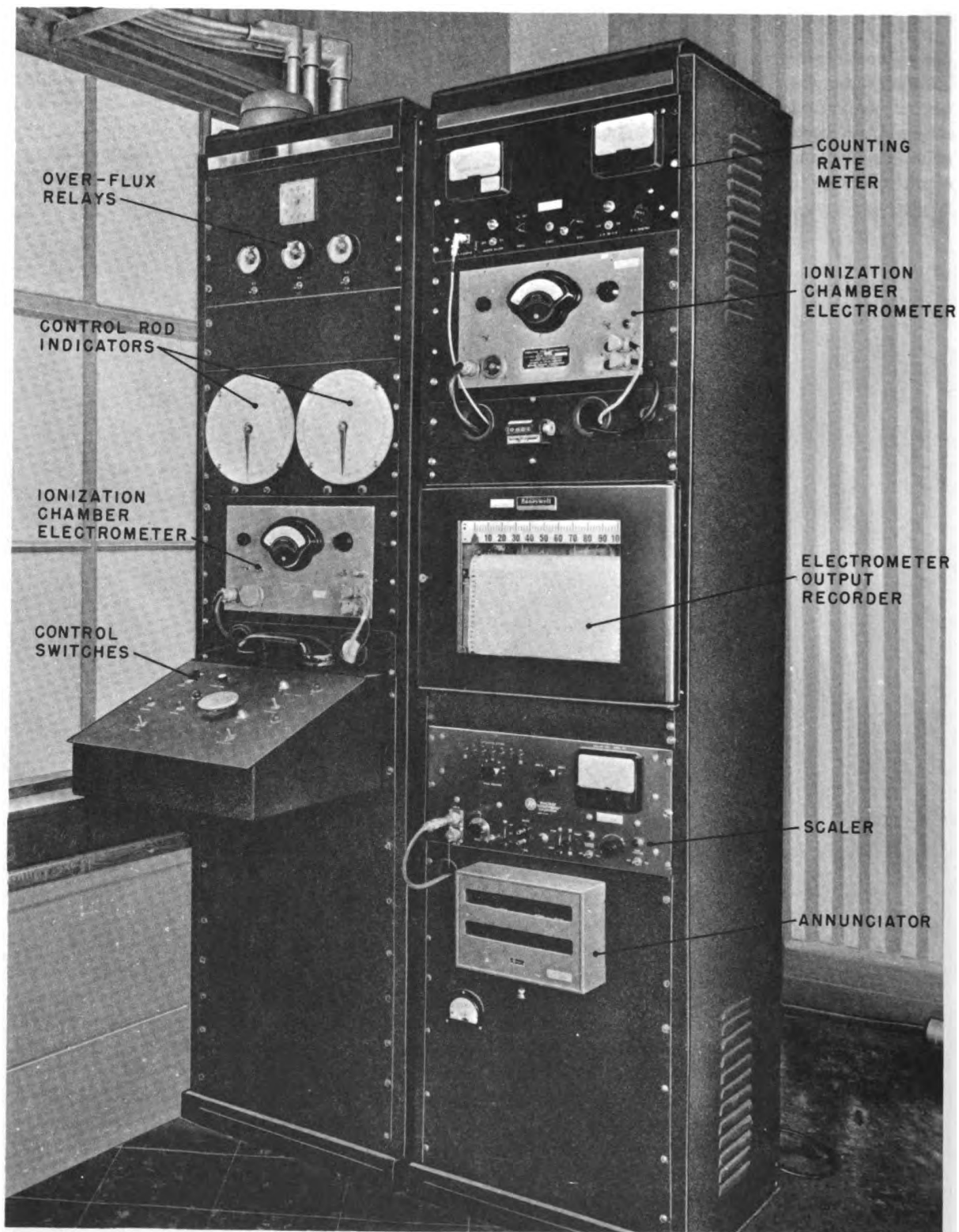


Fig. 1-9 Control console.

Operating Characteristics. The reactor is started into operation with a 250-millicurie Ra-Be neutron source inserted in the center of graphite stringer No. 4. This source emits about 3×10^6 neutrons/sec, which results in a shutdown power level of the reactor of about 0.5 mw. A very

the addition of the U^{235} . The changes in reactivity were observed by the multiplication of a 5.7-curie Po-Be neutron source inserted in the central exposure facility up to the center of the tube through the sphere. The measurements were made with suitably located BF_3 ionization

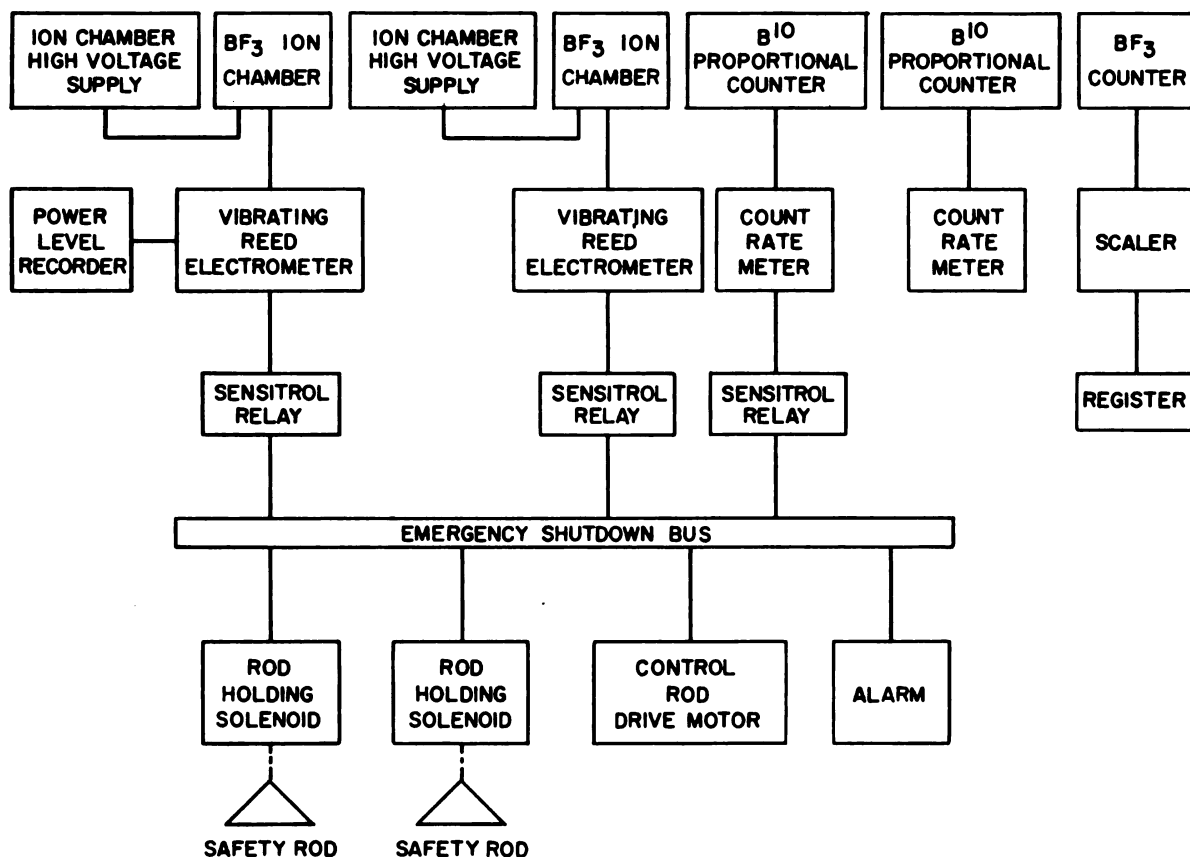


Fig. 1-10 Control instrumentation.

large part of the operation is carried out with a source in place in the reflector. This results in a stabilizing influence on the operation, particularly at the low power levels at which the reactor is operated.

Critical experiment. The critical experiment was performed by making successive additions of highly concentrated uranyl nitrate solution to the reactor core. Before the addition of any fissionable material, 11,980 cm³ of distilled water was added to the stainless-steel sphere, and the fuel system was flushed with helium. After each addition of fissionable material, measurements were made of the change in reactivity produced by

chambers, boron-lined neutron counters, and indium foils (approximately 90 mg/cm²) located in the graphite reflector, along a radial line from the edge of the sphere out to the edge of the graphite. Plots of the reciprocal counting rates as functions of the amount of fuel in the sphere permitted an extrapolation to the critical mass some time prior to the attainment of the critical state. Typical results are shown in Fig. 1-12, and from these the critical mass with all the control rods retracted is found to be 633.9 g of U^{235} .

A total of 638.2 g of U^{235} was added to the reactor core. This amount of fissionable material was sufficient to give a divergent chain reaction

with a period of 26 sec, corresponding to an excess reactivity of 0.21 per cent. A mass coefficient of reactivity for the WBNS was determined to be 0.050 per cent per g of U^{235} . It is interesting to note here that the mass coefficient appears

determined mass coefficient of reactivity, one finds that the coarse rod controls 1.05 per cent in reactivity.

Control- and safety-rod effectiveness. The reactivity control of the safety rods individually, of

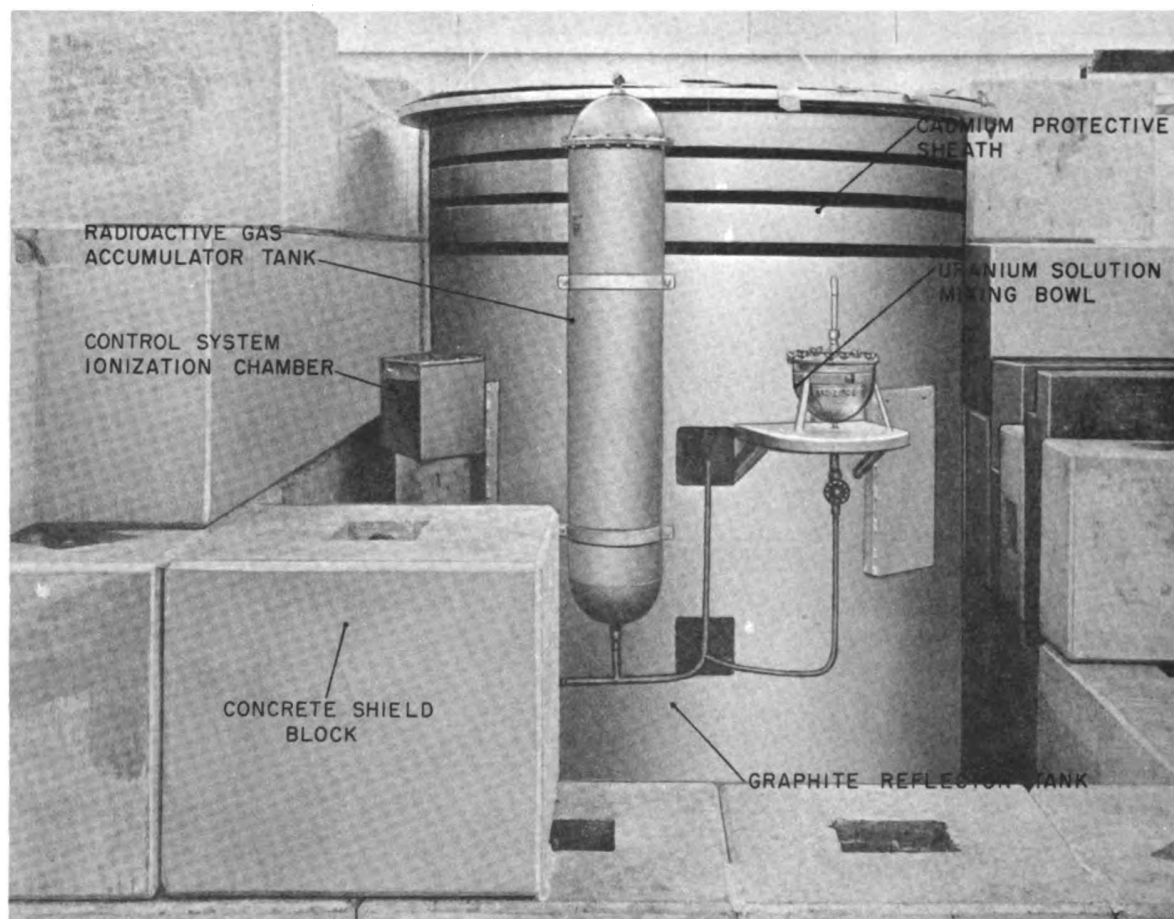


Fig. 1-11 Biological shield.

to be inversely proportional to the critical mass of the reactor, that is,

$$\left(\frac{\Delta k}{\Delta M}\right)_{\text{WBNS}} (Mc)_{\text{WBNS}} = 0.050 \times 633.9 = 31.7$$

During the critical assembly, measurements of the multiplication of the neutron source were also made with the coarse-control rod completely inside the reflector. Extrapolation of the reciprocal counting rate vs. amount of U^{235} for these data indicates the absorption of the coarse-control rod to be equivalent to 21 g of U^{235} . Using the afore-

the two rods together, and of the safety rods plus the coarse-control rod has been determined. Since the reactor loading and the amount of control by the coarse control are known, k_{eff} , the reproduction factor for the reactor, is known for the subcritical state produced with the safety rods withdrawn and the coarse-control rod completely in the reactor. The relative multiplications of a Po-Be neutron source inserted into the reactor core were determined with this configuration of control and safety rods and with other appropriate subcritical configurations necessary to determine

the amounts of control desired. The multiplication M in these subcritical states will be given by

$$M = \frac{1}{1 - k_{eff}}$$

Then, from two measurements of relative multiplications for different rod configurations, one obtains

$$(k_{eff})_1 = 1 - \frac{M_0}{M_1} [1 - (k_{eff})_0]$$

where M_0 and $(k_{eff})_0$ are the multiplication and reproduction factor for the known subcritical configuration, and M_1 is the measured multiplication for a configuration in which $(k_{eff})_1$ is unknown. After values of k_{eff} for these unknown configurations have been determined, the amount of control in the rods is readily obtainable.

The results of these measurements are shown in Table 1-1. It is to be noted that there is some

Table 1-1 Reactivity Control of Control and Safety Rods in WBNS

Rods	Reactivity control, %
Coarse-control rod.....	1.05
East safety rod.....	1.45
West safety rod.....	1.17
Coarse rod and east safety.....	2.44
Coarse rod and west safety....	1.93
East safety and west safety....	2.44
Coarse rod and both safety rods	3.27

shadowing among the rods, since in none of the cases is the amount of control by the two rods together equal to the sum of the amount of control by each rod separately. The two safety rods are identical in construction but are located in different regions of the reflector, so some difference in reactivity control by these rods should be expected.

Power calibration. The power level of the reactor has been determined by measuring the thermal-neutron flux in the central exposure facility. The power P of the reactor is given approximately by

$$P = \frac{N_{25}\sigma_{25}\bar{n}\bar{v}}{3 \times 10^{10}} \quad \text{watts}$$

where N_{25} is the number of atoms of U^{235} in the

sphere, σ_{25} is the fission cross section of U^{235} , and $\bar{n}\bar{v}$ is the average flux in the reactor core. From other measurements, the average flux $\bar{n}\bar{v}$ is determined to be 0.74 of the flux at the center of the spherical core. The neutron flux was measured with standardized indium foils that had been exposed to a known thermal flux. These measurements were then used to calibrate one of the BF_3

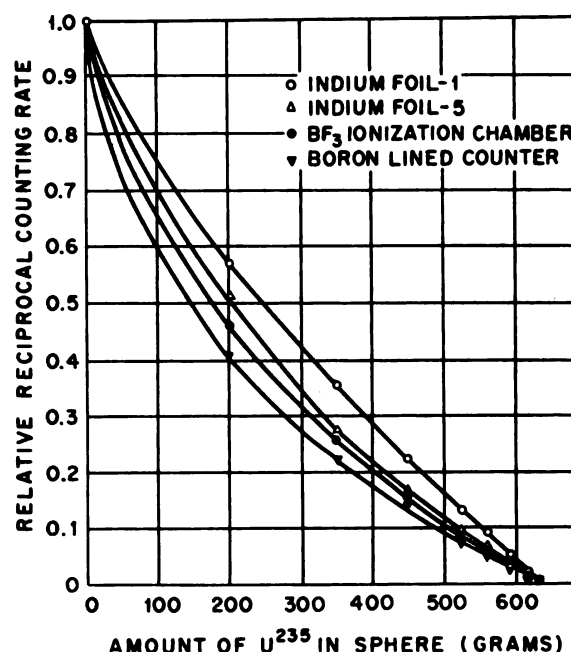


Fig. 1-12 Reciprocal counting rate obtained with three types of neutron detectors.

ionization chambers, and the recorder was used to monitor the power level of the WBNS. This absolute calibration is only accurate to about ± 30 per cent because of uncertainty in the standardization of the indium foils.

Gas evolution. Gas evolution resulting from the radiolysis of water in the fuel solution has been observed at operating levels as low as 0.1 watt. Experiments have been performed to determine the rate of hydrogen evolution and to identify the radioactive gases present in the accumulated gases.

The rate of gas evolution has been determined by the following procedures. After a period of operation in which the total fission energy release is determined from integration of the recorded

power level, the gas in the accumulator system is removed into a known volume in which the pressure and temperature are measured. Samples of the gas are then analyzed for hydrogen content

determinations are consistent, but this value for the rate of evolution is subject to the uncertainty in the power-level calibration previously mentioned. There are indications that the rate is

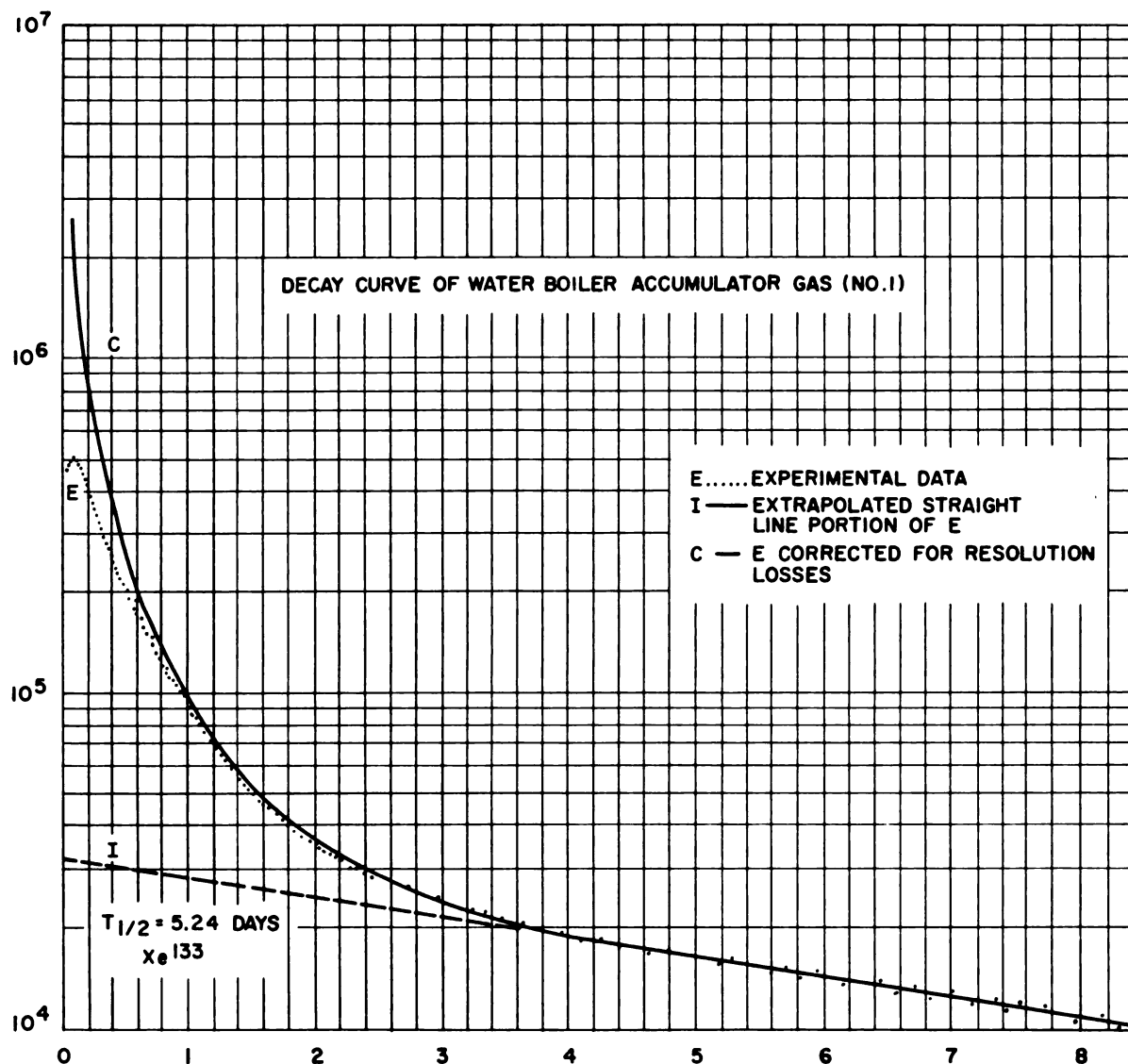


Fig. 1-13 Decay curve of water-boiler accumulator gas (No. 1).

by mass-spectrographic means. The rate of hydrogen evolution is then determined from the amount of hydrogen in the accumulator and the total operation for the period. Several measurements made after periods of operation totaling 30 to 50 watt-hr show the rate of hydrogen evolution to be approximately $10 \text{ cm}^3/\text{watt-hr}$. The

somewhat less (approximately 10 to 20 per cent) at power levels near 0.1 watt than at levels near 1 watt.

The sample to be tested for radioactive gases is transferred from the waste-disposal tank to an evacuated cylinder. The rate of decay is then followed by normal counting procedure and

plotted for visual observation. Only relative counting rates have been attempted.

The decay curves, as measured by a Geiger counter immersed in the cylinder, are shown in Figs. 1-13 and 1-14. The three gases which were

identified from the decay curves were Xe^{133} , Xe^{135} , and Kr^{88} . Figure 1-14 is obtained by subtracting the straight-line portion of the observed counting rate from the initial portion of the curve.

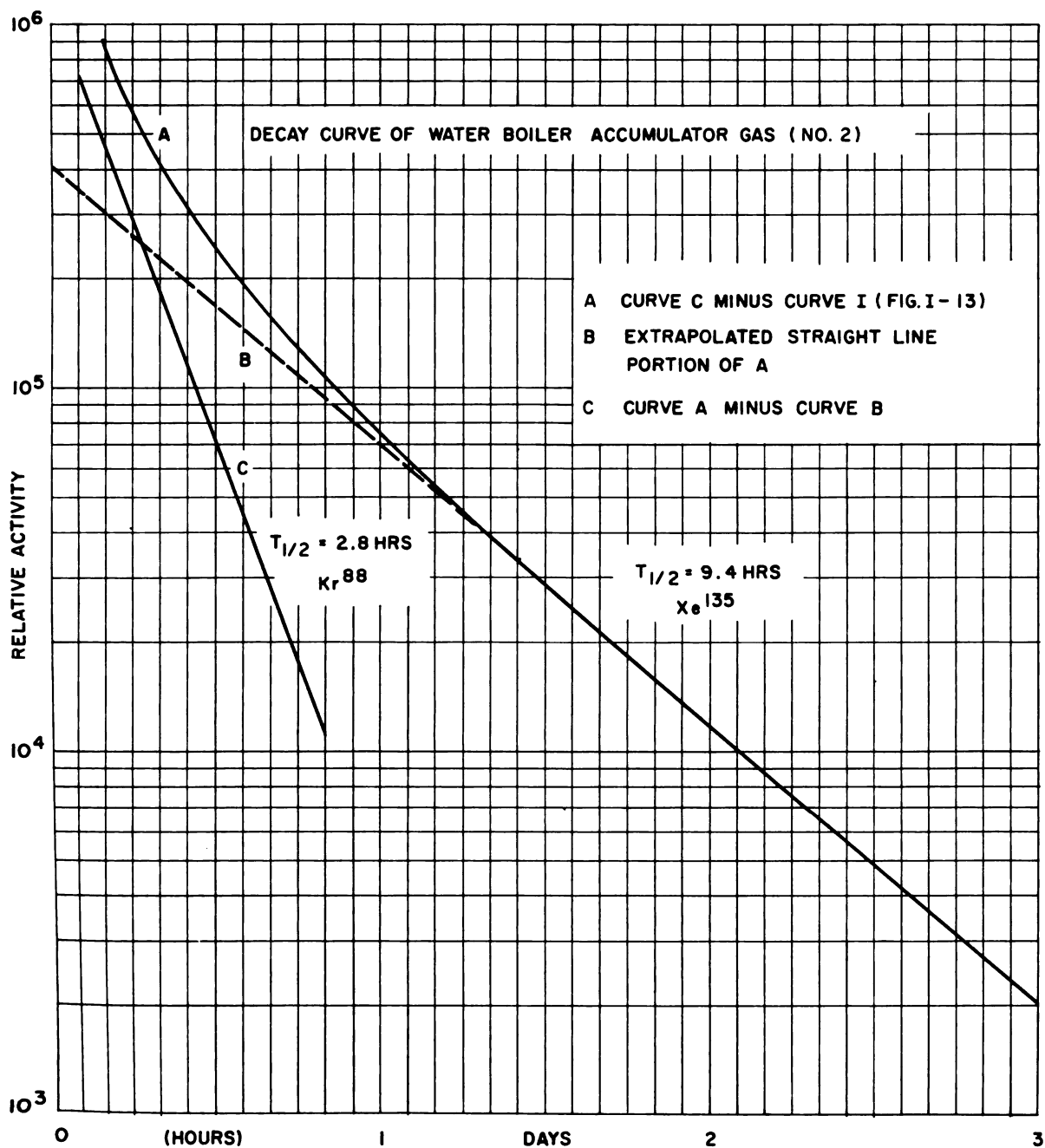
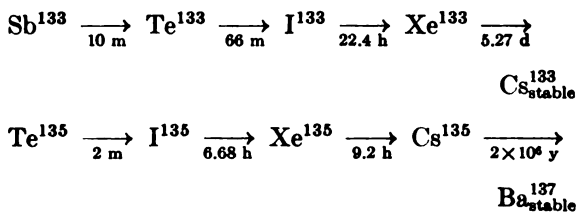


Fig. 1-14 Decay curve of water-boiler accumulator gas (No. 2).

In order to evaluate the relations found, the families of the fission-product gases that have been reported are shown below:



The yields of the xenon and iodine isotopes are presented in Table 1-2.

Table 1-2

Isotope	Fission yield, %	Independent yield, %
I^{133}	4.6	1.2
Xe^{133}	6.29	
I^{135}	5.6	
Xe^{135}	5.9	0.3

It can be seen that the time after the irradiation and the length of the irradiation will play an important part in the quantity of gases found in the accumulator. Present operating procedure is to flush out the accumulator when the total integrated power output is between 30 and 50 watt-hr. Since there is no fixed power level or time, it is difficult to reproduce any of the above-mentioned values. Some of the conditions which affect the equilibrium concentrations of fission-product gases in the accumulator are listed below:

1. Solubility of all elements except Xe and Kr gases.
2. Early adsorption of Br_2 and I_2 on metal surfaces (if emitted from the solution as gases; no evidence has been found that these gases are emitted from the WBNS core solution).
3. Sedimentation of metallic daughters in the static gas system.
4. Chemical effects of uranyl nitrate solution-water-decomposition products on Br_2 and I_2 molecules or their ions.
5. Choice of sampling time relative to the decay of the isotopes which have short half-lives.

If one is interested in the activity levels of a dynamic gas system, then other mechanisms—such as the choice of the wall material of the piping system, the flow rate of the gases, and the method of accumulating gases and particulate matter—must be considered in order to evaluate the system.

The activities identifiable by counting techniques and ionization-chamber measurement revealed only krypton and xenon isotopes. It is believed that these were probably the only radioactive gases present under the particular sampling conditions which prevailed.

Experimental Facilities. Thermal-neutron flux distributions in various available regions in the graphite reflector and in the central exposure facility have been measured by the activation of indium foils. Both bare and cadmium-covered indium foils were used at each point to give the cadmium ratio (defined by the ratio of bare-foil activity to cadmium-covered-foil activity) as well as the thermal-neutron flux in each region. Most of the data were taken with indium foils 1 cm^2 in area and approximately 90 mg/cm^2 in thickness. The cadmium covers were small boxes made of 20-mil cadmium sheet. For the flux distributions in the reflector, the foils were placed in a milled slot in the top of one of the graphite bars. This slot was just large enough for the foils to fit easily into it. The flux distribution in the central exposure facility was determined with foils mounted on a 2-SO aluminum foil holder. This resulted in some "neutron streaming" along this hole through the reflector. During each exposure a monitor foil was placed in a standard geometry in the reflector. The activities of these monitor foils permitted a suitable normalization of the activities of all the different foils.

The results of these measurements are shown in Figs. 1-15, 1-16, and 1-17, where the flux is normalized to unity at the center of the sphere in the central exposure facility in the sphere. The cadmium ratio with these indium foils varies from 3.7 in the central exposure facility to 36.0 at the edge of the reflector in stringer No. 1. The results show essentially symmetrical distributions about a vertical plane through the center of the core and graphite reflector, with a peaking of the flux inside the spherical core. The perturbations

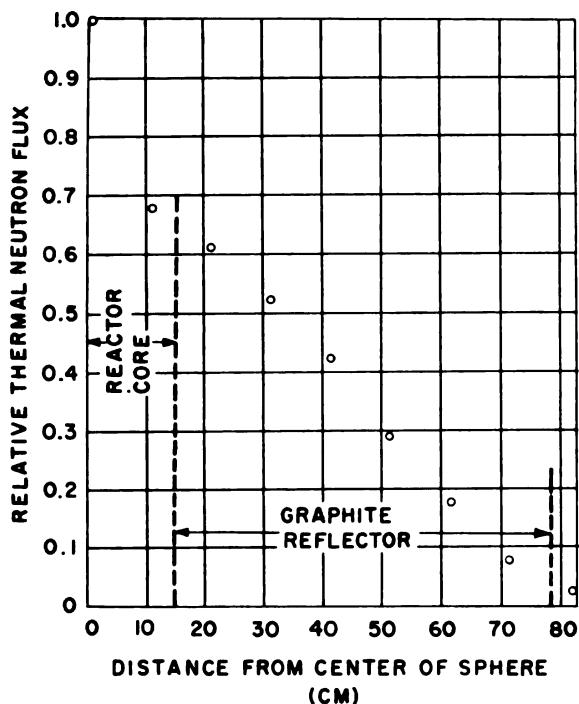


Fig. 1-15 Thermal-neutron flux distribution in central exposure facility.

introduced by the coarse-control rod and the east safety rod are shown in the flux distribution along the top of stringer No. 7 in Fig. 1-17. One should

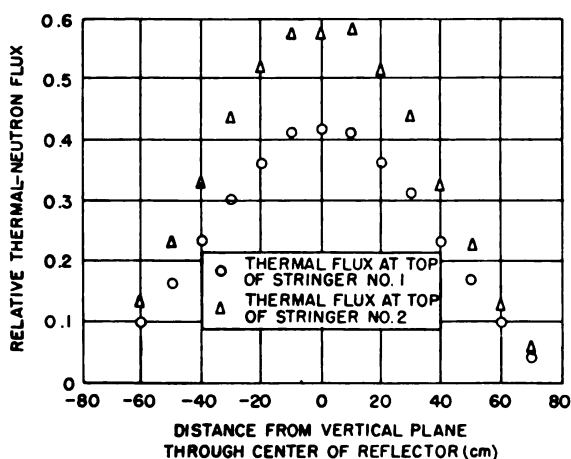


Fig. 1-16 Thermal-neutron flux distribution at top of removable stringers Nos. 1 and 2.

expect the distribution here to be very similar to that in stringer No. 2 (Fig. 1-16), and this is observed except for the depressions caused by the

two rods. The flattening by the coarse-control rod is somewhat greater than that by the east safety rod, since the safety rod was completely withdrawn from the reflector, and the contribution to the flux depression is that of the void from which the rod has been withdrawn. The coarse-control rod was withdrawn about 35 cm, which placed the end of the rod at approximately the outer edge of the stringer.

Cadmium-ratio measurements that have been made in the central exposure facility with foils of

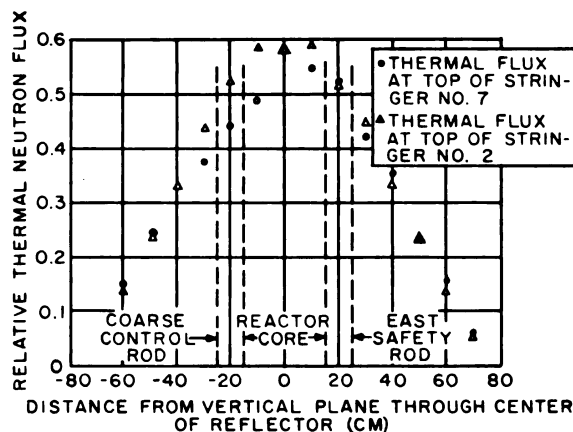


Fig. 1-17 Thermal-neutron flux distribution in removable stringers Nos. 2 and 7.

indium, gold, and manganese show that there is considerable fast-neutron flux in the core of the WBNS.

Fast-neutron Measurements in Thermal Column. Neutron measurements with a methane proportional counter have been made to estimate the flux of fast neutrons at the surface of the thermal column of a WBNS. This measurement is an attempt to identify specifically an energy-flux relationship at the position of interest. The prompt fast neutrons emitted from the sphere during fission have energies ranging from over 10 Mev to thermal values. In addition to the 2.5 neutrons per fission, there are prompt and delayed gamma radiations also emitted. These gamma radiations, plus those produced in the water by the $H^1(n,\gamma)H^2$ reaction and in the graphite by the $C^{12}(n,\gamma)C^{13}$ reaction, appear with a new spectrum of neutrons at the thermal-column surface, 3 ft from the central sphere.

The counting apparatus used in this investigation consisted of a methane-filled proton-recoil proportional chamber with an attached cathode-follower circuit which is connected to an amplifier-scaler. The methane chamber is a cadmium-shielded brass cylinder 2 in. in diameter and 15 in. long, filled with 70 cm (Hg pressure) of methane gas, with a 0.001-in.-diameter center-electrode wire. In the methane counter the proton-producing medium also serve as a proportional-counting

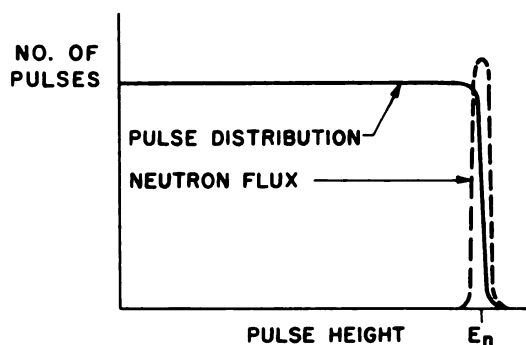


Fig. 1-18 Pulse number vs. height.

gas. Correct values of gas multiplication and discriminator settings allow counting of the recoil protons with gamma discrimination.

Because of the near equality of mass involved, the n,p scattering of neutrons by hydrogenous materials allows a fast-neutron detection method with unique characteristics. Since the scattering is isotropic, the recoil proton produced by the elastic collision of a neutron with a hydrogen nucleus will have, with equal probability, any energy up to the incident-neutron energy. If all the ionization produced by each proton is converted into a proportional pulse, examination of the pulse-height distribution obtained from a proton-recoil counter, along with knowledge of the n,p scattering cross section, will yield the incident-neutron energies. Although the proton range-energy relationships indicate difficulty in collecting all the ions produced by protons in the Mev range in any practical counting chamber, suitable wall corrections may be added to the response of an "infinite" chamber to allow interpretation of actual methane-chamber data.

If a monoenergetic beam of neutrons were to fall on a methane- or hydrogen-filled counting chamber in which wall effects, space-charge ef-

fects, and counting-geometry effects are assumed to be negligible, the resulting recoil-proton pulse-height distribution in terms of energy would be a constant value up to the incident-neutron energy E_n (Fig. 1-18). Beyond E_n there would be no pulses; that is, no recoils will have greater energy than the incident neutron. It may be seen from this energy-transfer mechanism that, if the differential pulse-height distribution is measured directly under these conditions (with a pulse-height analyzer, for example), the slope of the pulse distribution is proportional to the incident-neutron energy distribution.

A bias curve for the above theoretical counter with monoenergetic incident neutrons is shown in Fig. 1-19. If we determine the bias value B in terms of proton energy (Mev), then the bias curve will have the shape

$$\text{counts/min} = K \left(1 - \frac{B}{E_n} \right)$$

where K depends upon the chamber dimensions and the hydrogen-scattering cross section σ_s . The relation shows that when $B = 0$, all the available pulses will be counted, and as B increases to E_n , all the pulses are gradually "biased

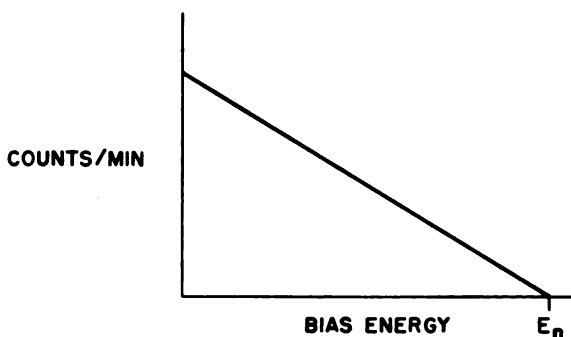


Fig. 1-19 Counts per minute vs. bias energy.

out" in a linear manner. Extending this relation to incident polyergic-neutron fluxes would give

$$\text{counts/min} = k\sigma_s E_n \left(1 - \frac{B}{E_n} \right)$$

where k is a constant. Because $\sigma_s E_n$ changes sharply with energy as shown in Table 1-3, the shape of the bias curve for polyergic neutrons will depend on the actual neutron energies and will generally show a greater response at lower ener-

gies. The cross section is fairly flat, however, above 2 Mev.

Table 1-3 Neutron-scattering Cross Section vs. Energy in Hydrogen

Neutron energy, Mev	σ_s , barns
0.05	16
0.10	14
0.50	8
1.0	4
5.0	1.7
10.0	0.8

In a methane counter of definite dimensions and pressure, corrections must be made for recoils produced which either strike the outer cylinder or pass out of the ends of the sensitive volume. Tables have been prepared which allow range equal to the longest dimension of the counting volume. In general, the addition of wall-effect corrections and scattering cross-section figures to the total counter response will result in an energy-response curve for constant bias similar to that shown in Fig. 1-20. The curve rises from

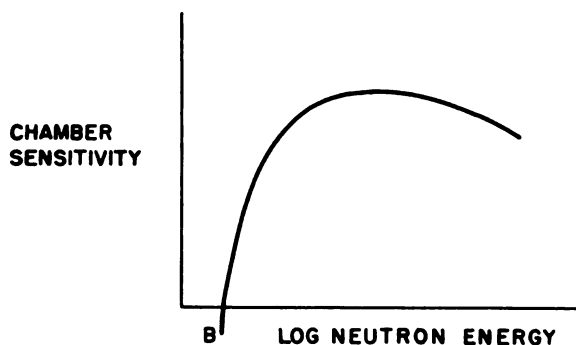


Fig. 1-20 Energy response.

zero at $E_n = B$ and reaches a maximum, after which the yield gradually decreases because of the change in σ_s .

From the above considerations, when the pulse-height distribution in a recoil chamber is accurately measured, derivation of the incident-neutron energy spectrum may be accomplished if (1) the pulse heights are known in terms of proton energy, (2) accurate wall-effect corrections are applied to the measured pulse-height distribution, and (3) the scattering cross-section values are taken into account.

In the past the methane counter has been used chiefly to compare neutron fluxes with the same energy distribution. In this measurement, bias curves were made for standard radium-beryllium and polonium-beryllium sources, and these known

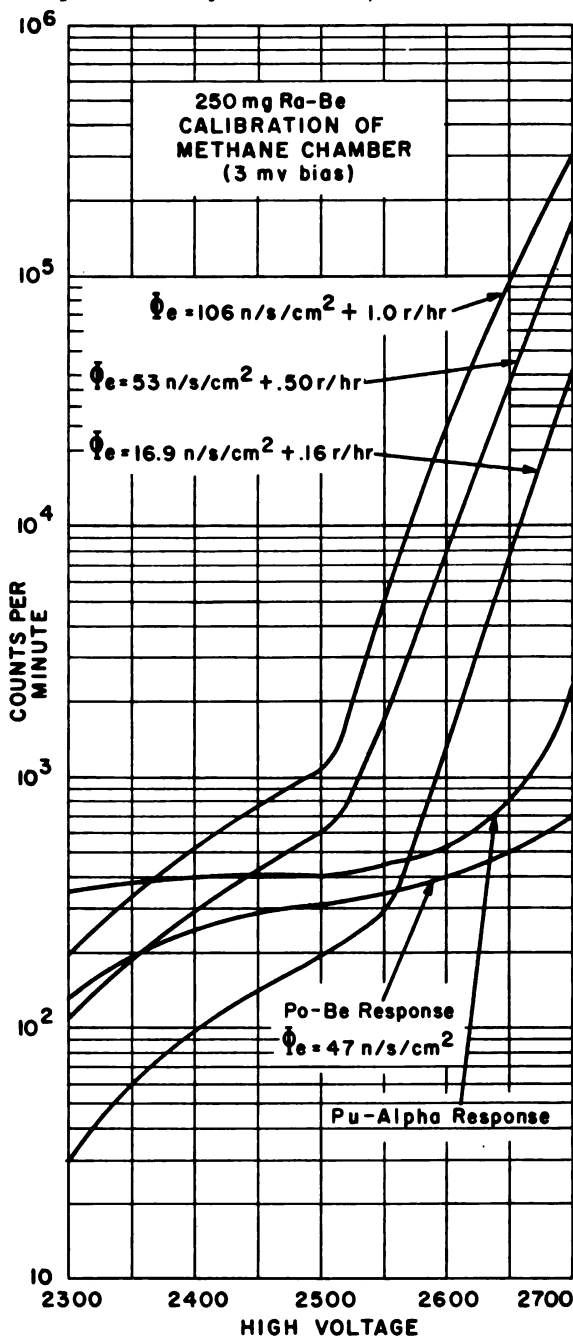


Fig. 1-21 250 mg Ra-Be calibrations of methane chamber, 3-mv bias.

fluxes were then used to derive sensitivity values for the bias values used. These curves are presented in Figs. 1-21, 1-22, and 1-23. The calibration sources used yielded neutron fluxes which required placing the source as close as 10 cm from

where x = source to end of sensitive volume distance, cm

L = length of sensitive volume, cm

In order to attach "bias-energy" values to the operating conditions, a needle with Pu^{239} (alpha

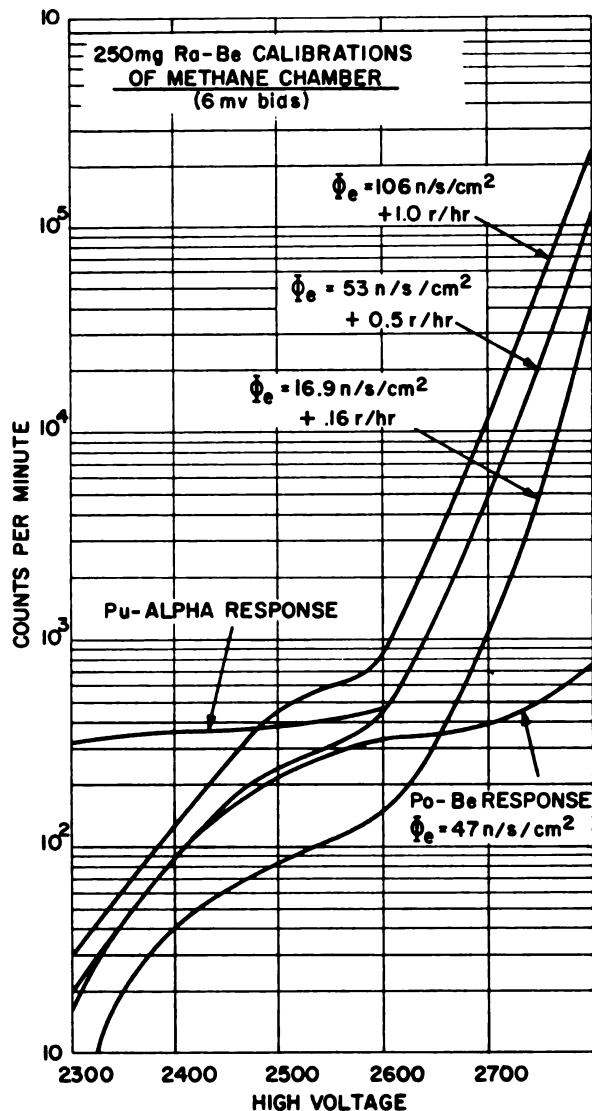


Fig. 1-22 250 mg Ra-Be calibrations of methane chamber, 6-mv bias.

the chamber tip. Under these conditions, the average flux in the sensitive volume appreciably differed from the flux at the chamber center. An effective flux Φ_e was calculated and used for the calibration

$$\Phi_e = \frac{\Phi_{at 1 cm}}{x^2 + xL}$$

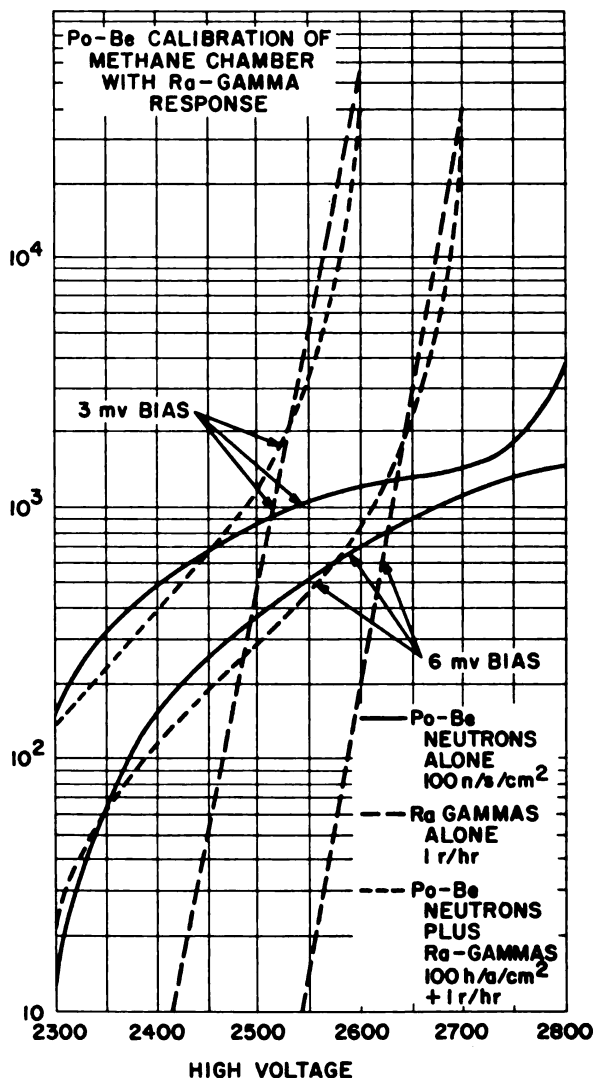


Fig. 1-23 Po-Be calibration of methane chamber with Ra-gamma response.

energy = 5.15 Mev) on the tip was introduced into the chamber. The resulting count rates are shown in Figs. 1-21 and 1-22. Pulse heights from the alpha ionization were measured with a calibrated oscilloscope and a gas amplification vs. voltage curve (Fig. 1-24) was obtained. The pulse height in the ionization region is ($A = 1$)

given by the equation

$$\text{Pulse height (volts)} = \frac{E_{a(\text{ev})}A}{32.5 \left(\frac{\text{volts}}{\text{ion pair}} \right) 6.28 \times 10^{18} \left(\frac{\text{electrons}}{\text{coulomb}} \right) C}$$

where C is the capacity of the center electrode and preamplifier input system ($6.2 \mu\text{f}$). Pulse heights

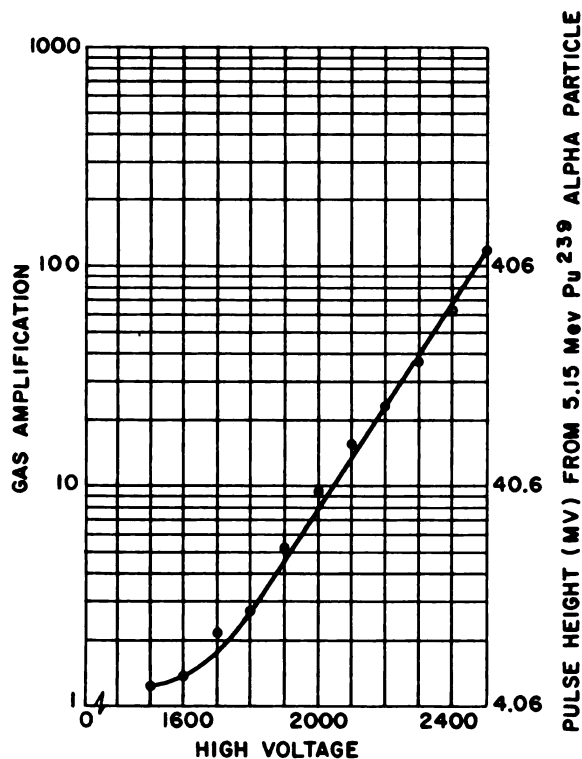


Fig. 1-24 Measured gas amplification of methane counter.

corresponding to the discriminator setting were measured using the oscilloscope and a pulse generator.

The original measurements were taken at the top surface of the cylindrical graphite moderator with the counter placed vertically to the central axis for several WBNS power levels (Fig. 1-25). Data obtained with the counter in this position are presented in Table 1-4. For purposes of interpretation, complete voltage curves were taken at several bias settings. These data are shown in Figs. 1-27 and 1-28.

Additional measurements were taken with the counter axis parallel to the top surface of the graphite to obtain attenuation factors through 12

in. of graphite (Fig. 1-26). Data obtained with and without the additional graphite are presented in Table 1-5. Attempts to obtain attenuation

Table 1-4 Fast-neutron Flux of WBNS above Surface of Thermal Column

Power, mw	Flux, n/cm ² /sec (average 2300-2400 volt values)	Flux/watt, n/cm ² /sec/watt
100	11.7 ± 3	117 ± 30
200	21.3 ± 3	106 ± 15
500	50.8 ± 13.4	102 ± 27
800	105 ± 21	131 ± 26
Avg.	114 ± 15

The deviations include errors in the instrument calibration and interpretation of counts.

Table 1-5 WBNS Fast-neutron Attenuation by 12.5 in. of Graphite at 500 mw

Probe voltage (3-mv bias)	Counting rates, cpm		Attenuation factors
	No graphite	Graphite	
2300	96	17.7	5.4
2400	297	80.5	3.7

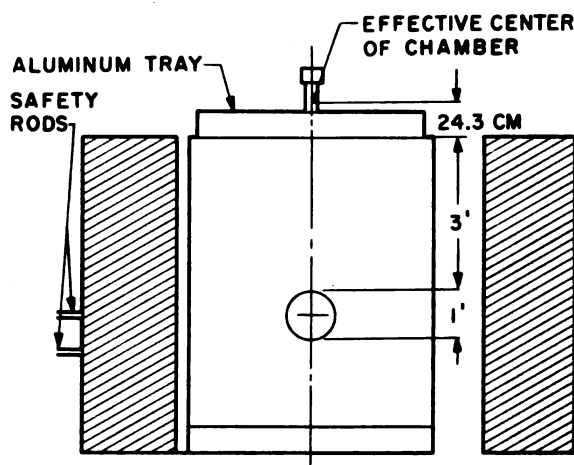


Fig. 1-25 Position of counter for fast-neutron flux measurement above thermal column.

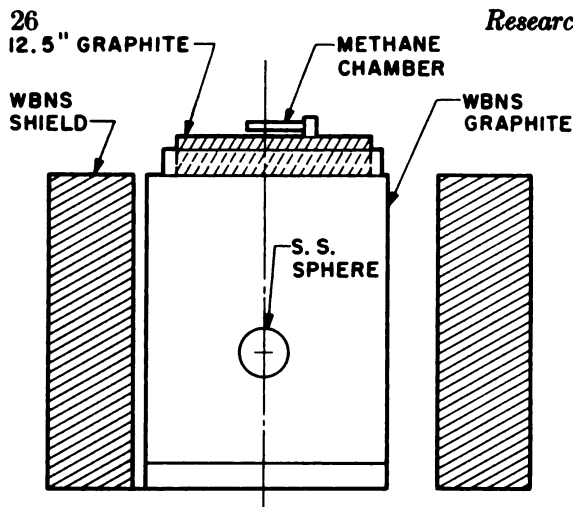


Fig. 1-26 Position of counter for obtaining fast-neutron attenuation factors.

factors for 1 and 2 in. of lead yielded data which are not interpretable (Table 1-6).

Table 1-6 WBNS Fast-neutron Attenuation by Lead Sheets at 500 mw

Probe voltage (3-mv bias)	Counting rates, cpm		
	0 in. lead	1 in. lead	2 in. lead
2300	...	71	79.5
2400	318	338	287
2500	644	576	574

Figure 1-29 shows the effect of removing the cadmium shield while the chamber was exposed to the WBNS flux.

Table 1-7 Calculated Bias Energies for Various Chamber Voltages and Discriminator Settings

Chamber voltage, volts	Bias energies, Mev	
	Discriminator = 3 mv	Discriminator = 6 mv
2300	0.089	0.187
2400	0.059	0.123
2500	0.033	0.070
2600	0.023	0.049
2700	0.015	0.031

Using the bias-energy values calculated during the calibration procedure, a sensitivity vs. neutron energy plot for the methane chamber was made for the 2300-volt 3-mv-bias operating condition. This response curve is shown in Fig. 1-30.

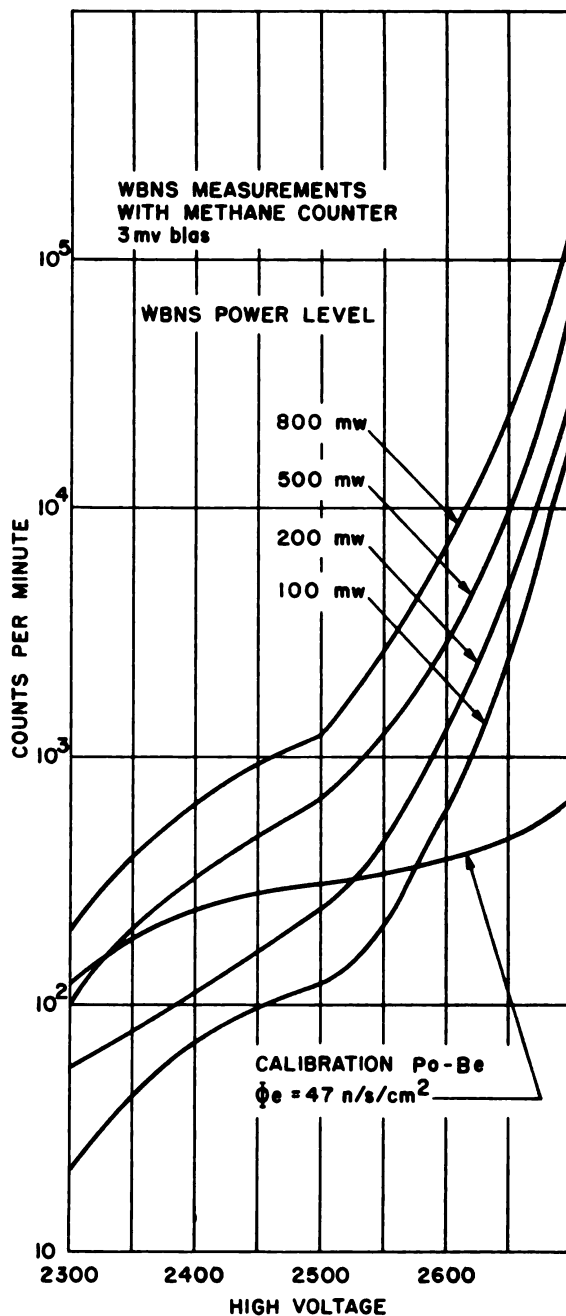


Fig. 1-27 WBNS measurements with methane counter.

Table 1-7 lists the bias energies for all the operating conditions used for the WBNS measurements.

It should be noted that the energy-response curve would simply consist of an asymptotic rise

gen, other reactions with the epithermal neutrons from the WBNS may be appreciable if heavier ionization than that due to photons is produced. The cadmium-shield effect (Fig. 1-29) shows that the chamber is sensitive to neutrons below the cadmium cutoff and that the capture gamma radiation from the cadmium has an effect. The count-

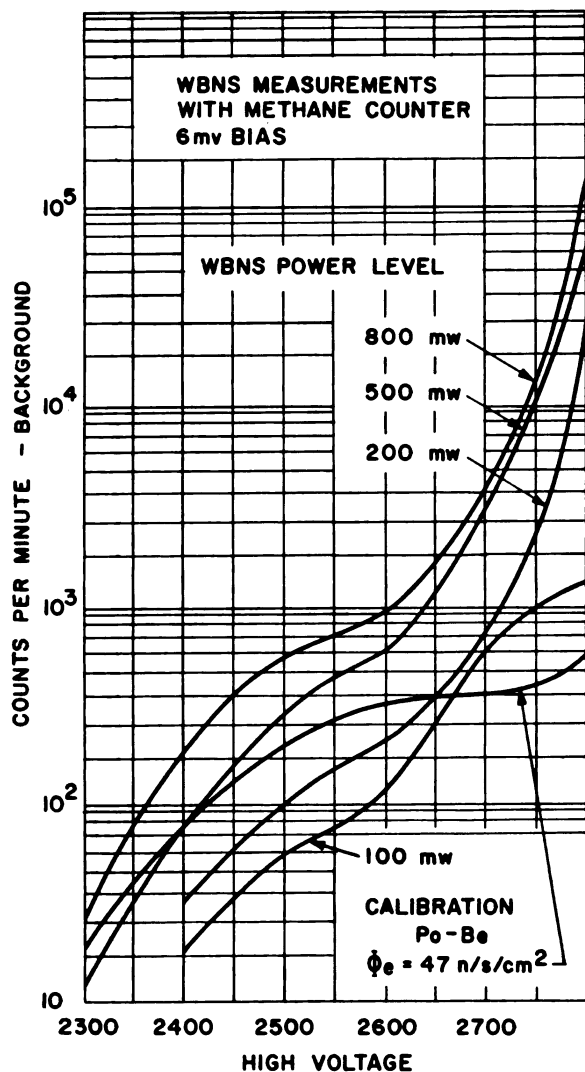


Fig. 1-28 WBNS measurements with methane counter, 6-mv bias.

to unity ($1 - B/E_n$) if the scattering cross section did not cause the gradual decrease after the maximum at about 0.2 Mev. Figure 1-31 shows calibration data for the Po-Be and the Ra-Be sources and the WBNS data at 500 mw plotted as a function of the calculated bias energies.

Although the energy-transfer mechanism for elastic collisions will not allow an appreciable contribution by recoils from nuclei other than hydro-

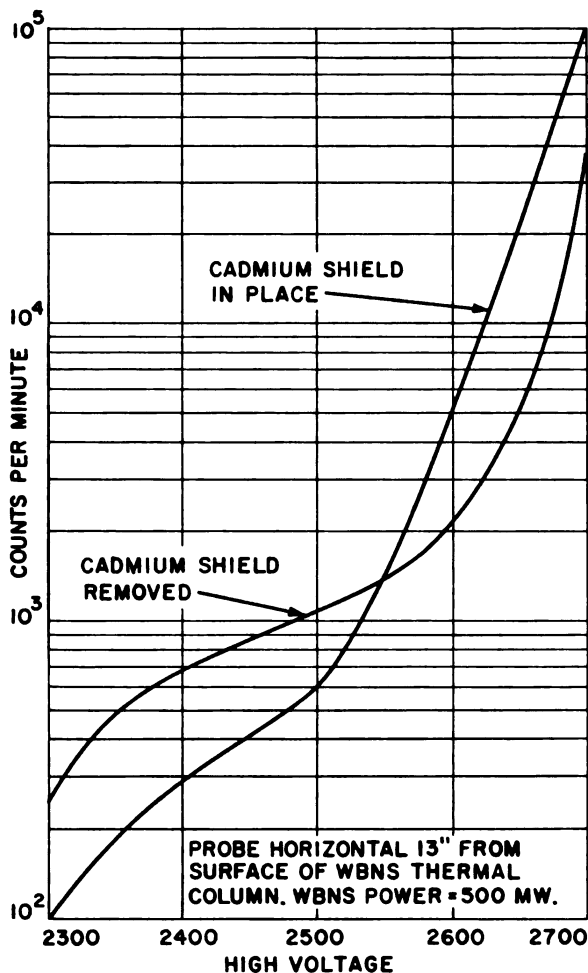


Fig. 1-29 Cadmium-shield effect on methane chamber, 6-mv bias.

ing rate vs. bias energy curve (Fig. 1-31) shows that, although the gamma field present for the Ra-Be curve is approximately 1 r/hr and the 500-mw WBNS gamma field is only about 0.15 r/hr, the WBNS curve nevertheless crosses the Ra-Be curve and remains at a higher value as the gamma-sensitive region is entered.

Because of the low neutron intensities involved in the measurement, it was necessary to use bias

values as close as possible to the gamma-sensitive region [i.e., the value of $(1 - B/E_n)$ approached unity]. The calculated values of the bias energies from Table 1-6 and the energy-response curve of Fig. 1-30 show that when the counter was used at the three levels (2300 volts, 3 mv; 2400 volts, 6 mv; and 2300 volts, 6 mv), the estimated flux

track plates. These data (Fig. 1-32) indicate that the fission spectrum from a solution-containing sphere similar to the WBNS reaches a maximum at a value slightly below 1 Mev. After moderation by 1 ft of graphite and 8.5 in. of bismuth, the moderated flux above 1 Mev is similar to the fission-spectrum distribution, but a much greater

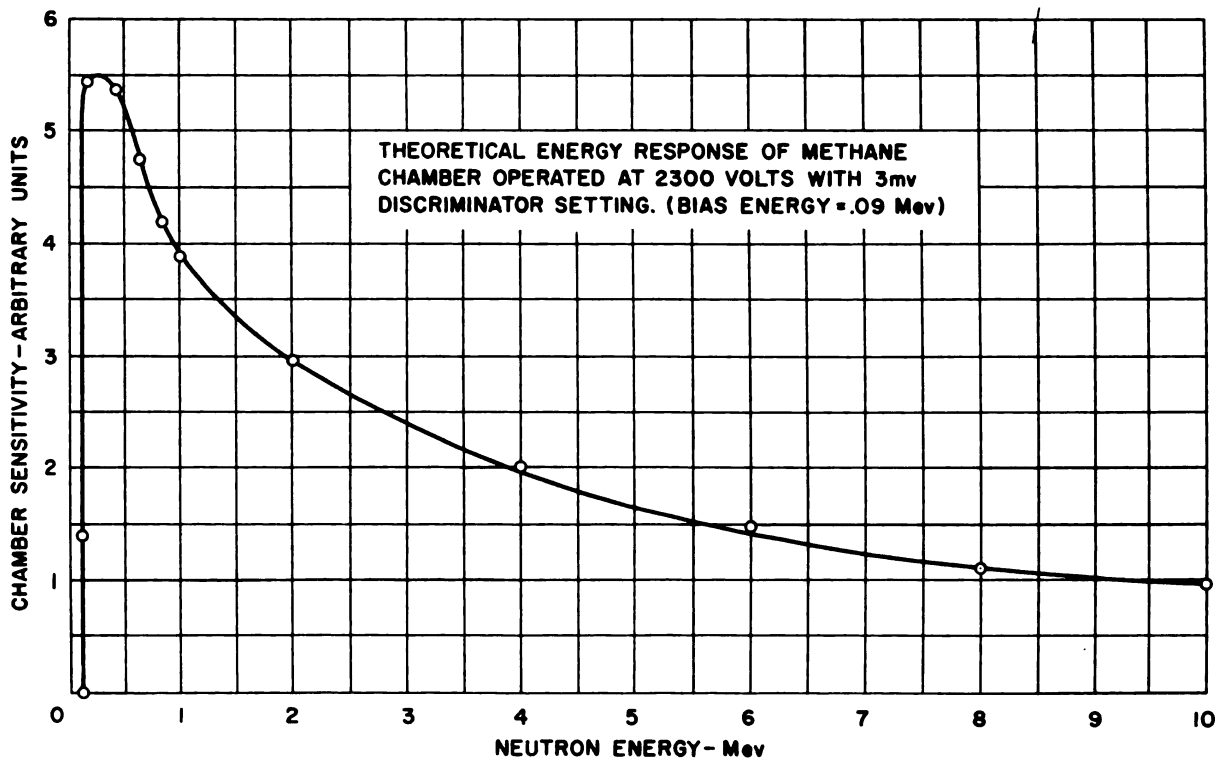


Fig. 1-30 Theoretical energy response of methane chamber operated at 2300 volts with 3-mv bias.

consisted of all neutron energies greater than 90, 132, and 187 kev, respectively.

Because counts of sufficient quantity to allow statistical accuracy could not be obtained for bias energies greater than the three gamma-free conditions, little information was gained revealing spectral differences between the calibration sources and the WBNS. Since the biologically significant fast-neutron spectra lie well above the lower limit of the counters' energy response (essentially above 0.5 Mev), the averaged flux values in Table 1-2 give a maximum measure of the fast-neutron dosage rate above the bias energies.

Investigations made of the fission spectrum both before and after moderation have been made at the Los Alamos water boiler using nuclear

relative flux is present below this energy. This is probably true of the WBNS.

Because the biological effect of fast-neutron radiation is largely due to energy-transfer phenomena dependent upon incident-neutron energy, accurate dosage-rate measurements require further data than are provided by determination of flux in neutrons per second per square centimeter. The Los Alamos data indicate that, while the biological effect of neutrons of energies greater than 1.0 Mev is large compared to neutrons having lesser energies, the WBNS neutron-flux spectrum may predominately exist below this energy. It is therefore necessary that the WBNS flux be measured as an energy function throughout the entire neutron spectrum and that the dosage rate be de-

terminated by summing the biological effect contributed in each energy increment.

Experimental Uses. The WBNS has been found extremely useful as a neutron-irradiation facility where a thermal flux of the order of 10^6 to 10^7 is

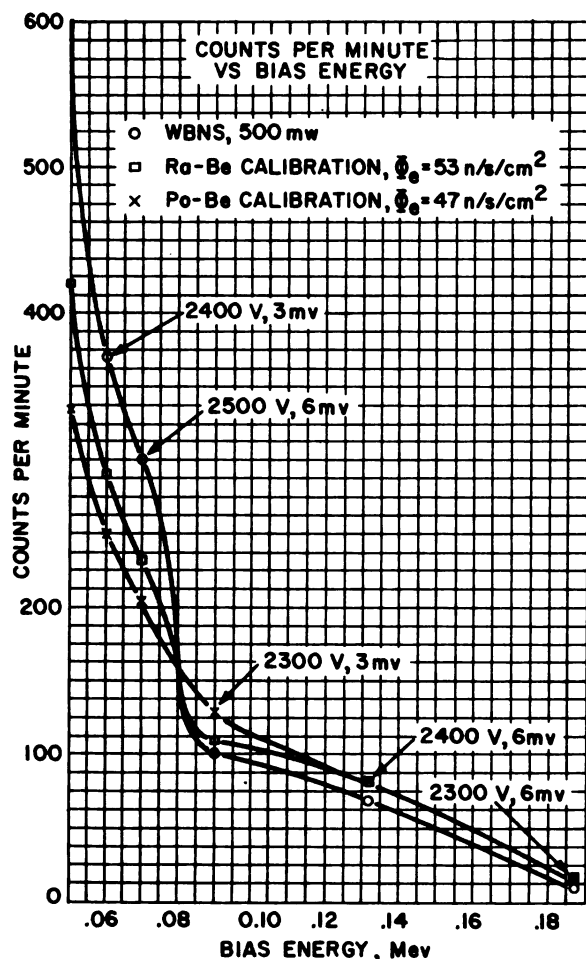


Fig. 1-31 Counts per minute vs. bias energy.

required. Subjects for irradiation can be placed in the central exposure facility or in one of the removable graphite stringers. There is also a flux of about 10^6 neutrons/cm²/sec/watt available at the top of the vertical thermal column. The calculated k_{∞} for the reactor is 1.56, and hence the leakage probability is 0.360, so that a large leakage flux per unit power is available for experimental purposes.

The various experiments in which the WBNS has already proved to be a valuable tool include:

1. Testing and calibration of ionization chambers, boron trifluoride proportional counters, and boron-lined counters for reactor instrumentation and neutron-physics studies.
2. Irradiation of foils to be used in the development of absolute counting techniques.
3. Irradiation of foils in the study of resonance absorption of neutrons in various elements.
4. Study of radiation effects in structure-sensitive materials.
5. Testing and calibration of health-physics instruments.
6. Testing of various materials for neutron-absorbing impurities by the danger-coefficient method to be described below.
7. Irradiation of iridium for research on electron-transfer isotopic-exchange reactions of iridium complex ions in aqueous solutions.

Of great importance has been the use of the WBNS as a neutron source for exponential experiments. The vertical thermal column provides a rather large extended source of thermal neutrons on which various subcritical assemblies can be constructed. Neutron-flux distributions are determined in these assemblies, which are mock-ups of the lattices in various types of reactors under study. Information derived from the measurements in the subcritical assemblies has contributed to the general knowledge of reactor theory and has aided materially in the design of specific reactor lattices.

The WBNS is useful as a producer of short-lived isotopes. By producing these isotopes in small quantities as needed, the number of elements available for tracer studies is increased, at the same time minimizing the problems of handling large levels of radioactive materials and allowing waste disposal by radioactive decay.

The formulas used for calculating the radioactivation by thermal neutrons are given below.

1. Definition of symbols:

- S = stable isotope of element
- N = radioactive isotope
- A = activity of N after irradiation
- t = irradiation time
- T = cooling time
- σ = thermal-neutron activation cross section of S
- f = thermal-neutron flux

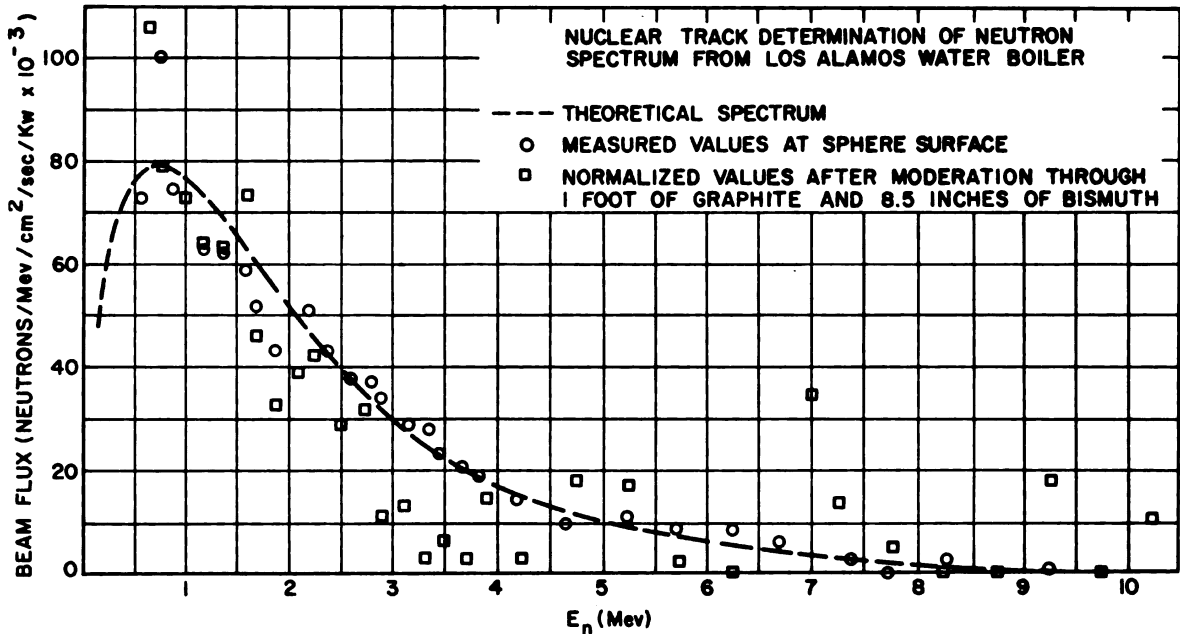


Fig. 1-32 Nuclear track determination of neutron spectrum from Los Alamos water boiler.

n = number of atoms of S
 k = natural abundance ratio of S

2. Single decay:

$$S \xrightarrow{\sigma} N \xrightarrow{\lambda}$$

$$A = \sigma f n k (1 - e^{-\lambda t}) e^{-\lambda t}$$

3. Two-chain decay:

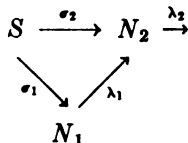
$$S \xrightarrow{\sigma} N_1 \xrightarrow{\lambda_1} N_2 \xrightarrow{\lambda_2}$$

$$A = \frac{\lambda_2 X}{\lambda_2 - \lambda_1} (e^{-\lambda_1 t} - e^{-\lambda_2 t}) + Y e^{-\lambda_2 t}$$

where $X = \sigma f n k (1 - e^{-\lambda_1 t})$

$$Y = \frac{\left\{ \lambda_2 \sigma f n k (e^{-\lambda_1 t} - e^{-\lambda_2 t}) + \sigma (\lambda_1 - \lambda_2) f n k (1 - e^{-\lambda_2 t}) \right\}}{\lambda_1 - \lambda_2}$$

4. Decay by isomeric transition:



$$A = \frac{\lambda_2 X}{\lambda_2 - \lambda_1} (e^{-\lambda_1 t} - e^{-\lambda_2 t}) + Y e^{-\lambda_2 t}$$

where $X = \sigma_1 f n k (1 - e^{-\lambda_1 t})$

$$Y = \frac{\left\{ \lambda_2 \sigma_1 f n k (e^{-\lambda_1 t} - e^{-\lambda_2 t}) + (\sigma_1 + \sigma_2) (\lambda_1 - \lambda_2) f n k (1 - e^{-\lambda_2 t}) \right\}}{\lambda_1 - \lambda_2}$$

The specific activity of every element immediately after irradiation has been calculated assuming a 1-hr irradiation in a flux of 4×10^7 neutrons/cm²/sec. All radioisotopes with a specific activity greater than 10 dis/sec/g and a half-life greater than 10 min have been tabulated in Table 1-8. These radioisotopes have been listed according to half-life in Table 1-9.

More than 60 elements with specific activities greater than 10 dis/sec/g can be activated in a 1-hr irradiation by the WBNS. Of the 100 radioisotopes that can be produced, 70 have half-lives short enough to permit disposal by radioactive decay.

In Table 1-10 a comparison has been made with the radioisotopes available for distribution from Oak Ridge National Laboratory. More than 40 radioisotopes are not available from ORNL because of their short half-life.

Short-half-life radioisotopes can be used to make accurate coincidence-loss determinations on

Table 1-8 Radioisotopes Produced by 1-watt Reactor

Radioisotope	Half-life	Produced by	Energy, Mev		Specific activity,* dis/sec/g
			β	γ	
Na ²⁴	14.8 hr	Na ²³ (n, γ)Na ²⁴	1.39	2.8, 1.4	25 \times 10 ³
Si ³¹	2.8 hr	Si ³⁰ (n, γ)Si ³¹	1.47	None	65 \times 10 ¹
P ³²	14.3 days	P ³¹ (n, γ)P ³²	1.70	None	31 \times 10 ¹
S ³⁵	87.1 days	S ³⁴ (n, γ)S ³⁵	0.17	None	25
Cl ³⁸	37 min	Cl ³⁷ (n, γ)Cl ³⁸	4.9, 1.1, 2.8	2.1, 1.6	64 \times 10 ³
A ⁴¹	1.83 hr	A ⁴⁰ (n, γ)A ⁴¹	1.2, 2.5	1.30	12 \times 10 ⁴
K ⁴²	12.4 hr	K ⁴¹ (n, γ)K ⁴²	3.6, 1.9	1.51	24 \times 10 ²
K ⁴⁰	1.2 \times 10 ⁹ years	nat occurring	1.36	1.46	11 \times 10 ¹
Sc ⁴⁶	85 days	Sc ⁴⁶ (n, γ)Sc ⁴⁶	1.5, 0.36	1.1, 0.89	66 \times 10 ²
Cr ⁵¹	28.5 days	Cr ⁵⁰ (n, γ)Cr ⁵¹	None	0.32 (K)	37 \times 10 ¹
Mn ⁵⁶	2.6 hr	Mn ⁵⁶ (n, γ)Mn ⁵⁶	2.9, 1.1, 0.7	2.1, 1.8, 0.85	13 \times 10 ⁶
Co ⁶⁰	5.9 years	Co ⁵⁹ (n, γ)Co ⁶⁰	0.31	1.2, 1.3	19 \times 10 ¹
Co ⁶⁰ (I.T.)	10.7 min	Co ⁵⁹ (n, γ)Co ⁶⁰	None	1.33	17 \times 10 ³
Ni ⁶⁵	2.5 hr	Ni ⁶⁴ (n, γ)Ni ⁶⁵	2.1, 1.0, 0.6	0.4, 1.1, 1.5	21 \times 10 ²
Cu ⁶⁴	12.8 hr	Cu ⁶³ (n, γ)Cu ⁶⁴	0.57 (58%)	1.34 (K)	60 \times 10 ³
Zn ⁶⁹	57 min	Zn ⁶⁸ (n, γ)Zn ⁶⁹	1.0	None	37 \times 10 ³
Zn ⁶⁹	14 hr	Zn ⁶⁸ (n, γ)Zn ⁶⁹	None	0.44	20 \times 10 ¹
Ga ⁷⁰	20.5 hr	Ga ⁶⁹ (n, γ)Ga ⁷⁰	1.65	None	27 \times 10 ⁴
Ga ⁷²	14.2 hr	Ga ⁷¹ (n, γ)Ga ⁷²	0.6 \rightarrow 3.15	0.6 \rightarrow 2.5	23 \times 10 ³
Ge ⁷⁵	82 min	Ge ⁷⁴ (n, γ)Ge ⁷⁵	1.10	None	28 \times 10 ³
Ge ⁷⁷	12 hr	Ge ⁷⁶ (n, γ)Ge ⁷⁷	2.20	0.50	28 \times 10 ¹
As ⁷⁶	26.8 hr	As ⁷⁵ (n, γ)As ⁷⁶	3.1, 2.5, 1.4, 0.4	0.6, 1.3, 1.8, 2.1	36 \times 10 ³
Se ⁸¹	18 min	Se ⁸⁰ (n, γ)Se ⁸¹	1.38	None	72 \times 10 ³
Se ⁸¹ (I.T.)	59 min	Se ⁸⁰ (n, γ)Se ⁸¹	None	0.10	22 \times 10 ²
Se ⁸³	25 min	Se ⁸² (n, γ)Se ⁸³	1.50	0.2, 0.4, 1.1	88
Br ⁸⁰	18 min	Br ⁷⁹ (n, γ)Br ⁸⁰	20	0.5	13 \times 10 ⁶
Br ⁸⁰ (I.T.)	4.4 hr	Br ⁷⁹ (n, γ)Br ⁸⁰	None	0.05, 0.04	72 \times 10 ³
Br ⁸²	35 hr	Br ⁸¹ (n, γ)Br ⁸²	0.46	0.5 \rightarrow 2.0	67 \times 10 ³
Kr ⁸⁶	4.4 hr	Kr ⁸⁴ (n, γ)Kr ⁸⁶	0.85	0.17, 0.37	24 \times 10 ³
Kr ⁸⁷	74 min	Kr ⁸⁶ (n, γ)Kr ⁸⁷	3.2	None	12 \times 10 ³
Rb ⁸⁶	19.5 days	Rb ⁸⁵ (n, γ)Rb ⁸⁶	1.8, 0.72	1.10	22 \times 10 ¹
Rb ⁸⁸	17.8 min	Rb ⁸⁷ (n, γ)Rb ⁸⁸	5.1, 3.3, 2.0	1.9, 0.9, 2.8	81 \times 10 ²
Rb ⁸⁷	6 \times 10 ¹⁰ years	nat occurring	0.27	0.05, 0.10	26 \times 10 ²
Y ⁹⁰	62 hr	Y ⁸⁹ (n, γ)Y ⁹⁰	2.18	None	36 \times 10 ²
Zr ⁹⁷	17 hr	Zr ⁹⁶ (n, γ)Zr ⁹⁷	1.91	0.80	84
Nb ⁹⁷	72 min	Zr ⁹⁶ (n, γ)Zr ⁹⁷ $\beta \rightarrow$ Nb ⁹⁷	1.27	0.66	24
Mo ⁹⁹	67 hr	Mo ⁹⁸ (n, γ)Mo ⁹⁹	1.2, 0.45	0.14, 0.76	12 \times 10 ¹
Mo ¹⁰¹	15 min	Mo ¹⁰⁰ (n, γ)Mo ¹⁰¹	2.1, 1.2	0.19, 0.96	43 \times 10 ²
Tc ¹⁰¹	15 min	Mo ¹⁰⁰ (n, γ)Mo ¹⁰¹ $\beta \rightarrow$ Tc ¹⁰¹	1.2	0.30	35 \times 10 ²
Ru ¹⁰³	41 days	Ru ¹⁰² (n, γ)Ru ¹⁰³	0.2, 0.7	0.3, 0.6	62
Ru ¹⁰⁶	4.5 hr	Ru ¹⁰⁴ (n, γ)Ru ¹⁰⁶	1.15	0.73	40 \times 10 ²
Pd ¹⁰⁹	13.2 hr	Pd ¹⁰⁸ (n, γ)Pd ¹⁰⁹	0.95	None	36 \times 10 ³
Pd ¹¹¹	26 min	Pd ¹¹⁰ (n, γ)Pd ¹¹¹	2.15	0.38	92 \times 10 ²
Ag ¹¹¹	7.5 days	Pd ¹¹⁰ (n, γ)Pd ¹¹¹ $\beta \rightarrow$ Ag ¹¹¹	1.06	0.34	56

* Immediately after a 1-hr irradiation.

Table 1-8 Radioisotopes Produced by 1-watt Reactor (Continued)

Radioisotope	Half-life	Produced by	Energy, Mev		Specific activity,* dis/sec/g
			β	γ	
Ag ¹¹⁰	270 days	Ag ¹⁰⁹ (n, γ)Ag ¹¹⁰	0.09, 0.53	0.65 \rightarrow 1.5	33
Cd ¹¹⁵	2.33 days	Cd ¹¹⁴ (n, γ)Cd ¹¹⁵	1.1, 0.46	0.52, 0.34	84 $\times 10^1$
Cd ¹¹⁷	2.8 hr	Cd ¹¹⁶ (n, γ)Cd ¹¹⁷	1.50	1.20	48 $\times 10^2$
In ¹¹⁷	1.9 hr	Cd ¹¹⁶ (n, γ)Cd ¹¹⁷ $\beta \rightarrow$ In ¹¹⁷	1.73	0.16	65 $\times 10^1$
In ¹¹⁴	48 days	In ¹¹³ (n, γ)In ¹¹⁴	1.98	0.19	31 $\times 10^1$
In ¹¹⁶	54 min	In ¹¹⁵ (n, γ)In ¹¹⁶	1.0	0.14 \rightarrow 2.1	16 $\times 10^6$
Sn ¹²¹	28 hr	Sn ¹²⁰ (n, γ)Sn ¹²¹	0.38	None	23 $\times 10^1$
Sn ¹²²	40 min	Sn ¹²² (n, γ)Sn ¹²²	1.26	0.15	96 $\times 10^1$
Sn ¹²⁵	10 min	Sn ¹²⁴ (n, γ)Sn ¹²⁵	2.4	None	23 $\times 10^2$
Sb ¹²²	2.8 days	Sb ¹²¹ (n, γ)Sb ¹²²	1.4, 1.9	0.57	80 $\times 10^2$
Sb ¹²⁴	60 days	Sb ¹²³ (n, γ)Sb ¹²⁴	0.5 \rightarrow 2.4	0.1 \rightarrow 2.3	10 $\times 10^1$
Sb ¹²⁴ (I.T.)	21 min	Sb ¹²³ (n, γ)Sb ¹²⁴	None	0.02	22 $\times 10^2$
Te ¹²⁷	9.3 hr	Te ¹²⁶ (n, γ)Te ¹²⁷	0.70	None	21 $\times 10^2$
Te ¹²⁹	72 min	Te ¹²⁸ (n, γ)Te ¹²⁹	1.75	0.3, 0.8	26 $\times 10^2$
Te ¹³¹	25 min	Te ¹³⁰ (n, γ)Te ¹³¹	1.80	2.20	11 $\times 10^2$
I ¹³¹	8 days	Te ¹³⁰ (n, γ)Te ¹³¹ $\beta \rightarrow$ I ¹³¹	0.6, 0.3	0.72, 0.36	26
I ¹³²	25 min	I ¹²⁷ (n, γ)I ¹³²	2.2, 1.6	0.43	11 $\times 10^6$
Xe ¹³³	5.3 days	Xe ¹³² (n, γ)Xe ¹³³	0.34	0.08	55
Xe ¹³³ (I.T.)	2.3 days	Xe ¹³² (n, γ)Xe ¹³³	None	0.23	33 $\times 10^2$
Xe ¹³⁵	9.2 hr	Xe ¹³⁴ (n, γ)Xe ¹³⁵	0.90	0.25	48 $\times 10^2$
Xe ¹³⁵ (I.T.)	15 min	Xe ¹³⁴ (n, γ)Xe ¹³⁵	None	0.52	16 $\times 10^5$
Cs ¹³⁴	2.3 years	Cs ¹³³ (n, γ)Cs ¹³⁴	0.66, 0.09	0.80, 0.60	16 $\times 10^1$
Ba ¹³⁹	85 min	Ba ¹³⁸ (n, γ)Ba ¹³⁹	2.27	1.0, 0.16	24 $\times 10^5$
La ¹⁴⁰	40 hr	La ¹³⁹ (n, γ)La ¹⁴⁰	1.3, 1.7, 2.3	0.09 \rightarrow 2.9	25 $\times 10^5$
Ce ¹⁴¹	28 days	Ce ¹⁴⁰ (n, γ)Ce ¹⁴¹	0.41, 0.56	0.14, 0.32	48
Ce ¹⁴³	33 hr	Ce ¹⁴² (n, γ)Ce ¹⁴³	1.1, 1.37	0.29, 0.06	38 $\times 10^1$
Pr ¹⁴³	19.3 hr	Pr ¹⁴¹ (n, γ)Pr ¹⁴³	2.1, 0.66	1.6, 0.6, 0.5	67 $\times 10^6$
Nd ¹⁴⁷	11 days	Nd ¹⁴⁶ (n, γ)Nd ¹⁴⁷	0.80, 0.36	0.09, 0.53	11 $\times 10^1$
Nd ¹⁴⁹	1.7 hr	Nd ¹⁴⁸ (n, γ)Nd ¹⁴⁹	0.9, 1.5	0.03 \rightarrow 0.65	76 $\times 10^2$
Nd ¹⁵¹	12 min	Nd ¹⁵⁰ (n, γ)Nd ¹⁵¹	1.9	0.08 \rightarrow 1.14	17 $\times 10^4$
Pm ¹⁴⁹	47 hr	Nd ¹⁴⁸ (n, γ)Nd ¹⁴⁹ $\beta \rightarrow$ Pm ¹⁴⁹	1.05	1.3, 0.28	64
Pm ¹⁵¹	27 hr	Nd ¹⁵⁰ (n, γ)Nd ¹⁵¹ $\beta \rightarrow$ Pm ¹⁵¹	1.1	0.71, 0.34	34 $\times 10^2$
Sm ¹⁵³	47 hr	Sm ¹⁵² (n, γ)Sm ¹⁵³	0.7, 0.8	0.07, 0.1, 0.6	94 $\times 10^5$
Sm ¹⁵⁵	25 min	Sm ¹⁵⁴ (n, γ)Sm ¹⁵⁵	1.8	0.25, 0.11	17 $\times 10^1$
Eu ¹⁵²	9.2 hr	Eu ¹⁵¹ (n, γ)Eu ¹⁵²	1.88	1.2	80 $\times 10^5$
Eu ¹⁵²	5.3 years	Eu ¹⁵¹ (n, γ)Eu ¹⁵²	0.9	0.3	28 $\times 10^2$
Eu ¹⁵⁴	5.3 years	Eu ¹⁵³ (n, γ)Eu ¹⁵⁴	1.9, 0.7, 0.3	1.2	12 $\times 10^2$
Gd ¹⁵⁰	18 hr	Gd ¹⁴⁹ (n, γ)Gd ¹⁵⁰	0.90	0.35	59 $\times 10^2$
Tb ¹⁶⁰	75 days	Tb ¹⁵⁹ (n, γ)Tb ¹⁶⁰	0.9, 0.5, 0.4	0.1 \rightarrow 1.0	26 $\times 10^2$
Dy ¹⁶⁵	2.3 years	Dy ¹⁶⁴ (n, γ)Dy ¹⁶⁵	1.3, 0.9, 0.4	0.09, 0.4, 0.8	56 $\times 10^6$
Ho ¹⁶⁶	27 hr	Ho ¹⁶⁵ (n, γ)Ho ¹⁶⁶	1.84	1.36	22 $\times 10^4$
Er ¹⁷¹	7.5 hr	Er ¹⁷⁰ (n, γ)Er ¹⁷¹	1.5, 1.0, 0.7	0.8, 0.3	50 $\times 10^5$

* Immediately after a 1-hr irradiation.

Table 1-8 Radioisotopes Produced by 1-watt Reactor (Continued)

Radioisotope	Half-life	Produced by	Energy, Mev		Specific activity,* dis/sec/g
			β	γ	
Tm ¹⁷⁰	129 days	Tm ¹⁶⁹ (n, γ)Tm ¹⁷⁰	0.97, 0.89	0.08	38×10^2
Yb ¹⁶⁹	32 days	Yb ¹⁶⁹ (n, γ)Yb ¹⁶⁹	None	0.11 (K)	22×10^2
Yb ¹⁷⁶	4.1 days	Yb ¹⁷⁴ (n, γ)Yb ¹⁷⁶	0.5, 0.13	0.35	14×10^3
Yb ¹⁷⁷	2.0 hr	Yb ¹⁷⁶ (n, γ)Yb ¹⁷⁷	1.30	0.15	28×10^3
Lu ¹⁷⁷	6.8 days	Yb ¹⁷⁶ (n, γ)Yb ¹⁷⁷ $\beta \rightarrow$ Lu ¹⁷⁷	0.5, 0.4, 0.2	0.1, 0.2, 0.3	67
Lu ¹⁷⁷	6.8 days	Lu ¹⁷⁶ (n, γ)Lu ¹⁷⁷	0.5, 0.4, 0.2	0.1, 0.2, 0.3	56×10^3
Lu ¹⁷⁸	3.7 hr	Lu ¹⁷⁶ (n, γ)Lu ¹⁷⁸	1.2, 1.1	0.09	80×10^4
Hf ¹⁸¹	46 days	Hf ¹⁸⁰ (n, γ)Hf ¹⁸¹	0.41	0.34, 0.48	30×10^1
Ta ¹⁸²	113 days	Ta ¹⁸¹ (n, γ)Ta ¹⁸²	1.1, 0.53	1.3, 0.22, 0.16	67×10^1
W ¹⁸⁶	77 days	W ¹⁸⁴ (n, γ)W ¹⁸⁶	0.43	None	32
W ¹⁸⁷	25 hr	W ¹⁸⁶ (n, γ)W ¹⁸⁷	1.35, 0.60	0.7, 0.5, 0.08	36×10^3
Re ¹⁸⁶	90 hr	Re ¹⁸⁵ (n, γ)Re ¹⁸⁶	1.1, 0.93	0.1, 0.8, 0.6	37×10^3
Re ¹⁸⁸	18 hr	Re ¹⁸⁷ (n, γ)Re ¹⁸⁸	2.1	0.2, 0.5, 1.4	23×10^4
Os ¹⁹¹	15 days	Os ¹⁹⁰ (n, γ)Os ¹⁹¹	0.14	0.13, 0.04	51×10^1
Os ¹⁹²	32 hr	Os ¹⁹² (n, γ)Os ¹⁹²	1.1	1.2	18×10^2
Ir ¹⁹²	70 days	Ir ¹⁹¹ (n, γ)Ir ¹⁹²	0.70	0.1 \rightarrow 1.0	19×10^3
Ir ¹⁹⁴	19 hr	Ir ¹⁹³ (n, γ)Ir ¹⁹⁴	2.1, 0.48	1.43, 0.38	36×10^4
Pt ¹⁹⁷	18 hr	Pt ¹⁹⁶ (n, γ)Pt ¹⁹⁷	0.70	0.07, 0.19	55
Pt ¹⁹⁹	31 min	Pt ¹⁹⁸ (n, γ)Pt ¹⁹⁹	1.80	None	26×10^3
Au ¹⁹⁹	3.2 days	Pt ¹⁹⁸ (n, γ)Pt ¹⁹⁹ Au ¹⁹⁹	0.32	0.20, 0.23	14×10^1
Au ¹⁹⁸	2.7 days	Au ¹⁹⁷ (n, γ)Au ¹⁹⁸	0.97	0.41	13×10^4
Hg ¹⁹⁷	65 hr	Hg ¹⁹⁶ (n, γ)Hg ¹⁹⁷	None	0.20 (K)	50×10^3
Hg ²⁰³	44 days	Hg ²⁰² (n, γ)Hg ²⁰³	0.20	0.28	70
Th ²³²	23.5 min	Th ²³² (n, γ)Th ²³²	1.23	None	64×10^4
Pa ²³³	27.4 days	Th ²³² (n, γ)Th ²³³ Pa ²³³	0.23, 0.53	0.03, 0.42	45×10^1

* Immediately after a 1-hr irradiation.

radioactive-measurement counting systems and may facilitate neutron-activation analysis by half-life identification.

Additional radioisotopes may be manufactured by fast-neutron n,p and n,α reactions since the reactor has a fast-neutron flux of 2×10^7 neutrons/cm²/sec.

The WBNS is also well suited for thermal-neutron-absorption cross-section measurements by the danger-coefficient technique. This technique involves the insertion of a neutron absorber in a reactor which produces a change in the reactivity. By properly calibrating a reactor control rod with the insertion of known amounts of neutron-absorbing material into the reactor core, the

displacement of the control rod can be used as a precision measure of the absorption cross sections. Since the total neutron-absorption cross section of the WBNS core is quite small (it has been calculated to be about 1440 cm²), the WBNS is very sensitive to the effects of neutron absorbers inserted in the central exposure facility.

Apparatus for this type of measurement with the WBNS consists of a BF₃ ionization chamber located between the shield and the reflector. The signal from the ion chamber is fed to a sensitive galvanometer (10⁻¹⁰ amp/mm), which is used as a null device. The current from the ionization chamber is balanced by current from a standard source consisting of a potentiometer and a set of

Table 1-9 Increasing Half-life

Half-life	Radio-isotope	Half-life	Radio-isotope
10. min	Sn ¹²⁵	1.0 days	W ¹⁸⁷
10.7 min	Co ⁹⁰	1.10 days	Pm ¹⁵¹
12. min	Nd ¹⁵¹	1.10 days	Ho ¹⁶⁶
15. min	Mo ¹⁰¹	1.12 days	As ⁷⁶
15. min	Tc ¹⁰¹	1.16 days	Sn ¹²¹
15. min	Xe ¹³⁵ (I.T.)	1.33 days	Os ¹⁹⁸
17.8 min	Rb ⁸⁸	1.40 days	Ce ¹⁴³
18. min	Se ⁸¹	1.45 days	Br ⁸²
18. min	Br ⁸⁰	1.65 days	La ¹⁴⁰
20. min	Ga ⁷⁰	2.0 days	Pm ¹⁴⁹
21. min	Sb ¹²⁴ (I.T.)	2.0 days	Sm ¹⁵²
23.5 min	Th ²³³	2.3 days	Xe ¹³³ (I.T.)
25. min	Se ⁸³	2.33 days	Cd ¹¹⁵
25. min	Te ¹³¹	2.6 days	Y ⁹⁰
25. min	I ¹²⁸	2.7 days	Au ¹⁹⁸
26. min	Pd ¹¹¹	2.7 days	Hg ¹⁹⁷
31. min	Pt ¹⁹⁹	2.8 days	Mo ⁹⁹
37. min	Cl ³⁸	2.8 days	Sb ¹²²
40. min	Sn ¹²³	3.0 days	Nb ⁹⁶
54. min	In ¹¹⁶	3.2 days	Au ¹⁹⁹
57. min	Zn ⁹⁹	3.75 days	Re ¹⁸⁶
59. min	Se ⁸¹ (I.T.)	4.1 days	Yb ¹⁷⁶
1.2 hr	Te ¹²⁹	5.3 days	Xe ¹³³
1.2 hr	Nb ⁹⁷	6.8 days	Lu ¹⁷⁷
1.24 hr	Kr ⁸⁷	7.5 days	Ag ¹¹¹
1.36 hr	Ge ⁷⁶	8.0 days	I ¹³¹
1.42 hr	Ba ¹³⁹	11.0 days	Nd ¹⁴⁷
1.7 hr	Nd ¹⁴⁹	14.3 days	P ³²
1.83 hr	A ⁴¹	15. days	Os ¹⁹¹
1.9 hr	In ¹¹⁷	19.5 days	Rb ⁹⁶
2. hr	Yb ¹⁷⁷	26.5 days	Cr ⁵¹
2.3 hr	Dy ¹⁶⁵	27.4 days	Pa ²³³
2.5 hr	Ni ⁶⁵	28. days	Ce ¹⁴¹
2.6 hr	Mn ⁶⁶	32. days	Yb ¹⁶⁹
2.8 hr	Si ³¹	41. days	Ru ¹⁰⁸
2.8 hr	Cd ¹¹⁷	43. days	Cd ¹¹⁶
3.7 hr	Lu ¹⁷⁶	44. days	Hg ²⁰³
4.4 hr	Kr ⁸⁶	46. days	Hf ¹⁸¹
4.4 hr	Br ⁸⁰ (I.T.)	48. days	In ¹¹⁴ (I.T.)
4.5 hr	Ru ¹⁰⁶	60. days	Sb ¹²⁴
7.5 hr	Er ¹⁷¹	70. days	Ir ¹⁹²
9.2 hr	Xe ¹³⁵	75. days	Tb ¹⁶⁰
9.2 hr	Eu ¹⁵²	77. days	W ¹⁸⁶
9.3 hr	Te ¹²⁷	85. days	Sc ⁴⁶
12. hr	Ge ⁷⁷	87.1 days	S ³⁵
12.4 hr	K ⁴²	113. days	Ta ¹⁸²
12.8 hr	Cu ⁶⁴	129. days	Tm ¹⁷⁰
13.2 hr	Pd ¹⁰⁸	270. days	Ag ¹¹⁰
14.0 hr	Zn ⁶⁹ (I.T.)	2.3 years	Ce ¹³⁴
14.2 hr	Ga ⁷²	5.3 years	Eu ¹⁵³
14.8 hr	Na ²⁴	5.3 years	Eu ¹⁵⁴
17. hr	Zr ⁹⁷	5.9 years	Co ⁶⁰
18. hr	Gd ¹⁴⁹	1.2 × 10 ⁹	
18. hr	Re ¹⁸⁸	years	K ⁴⁰
18. hr	Pt ¹⁹⁷	6 × 10 ¹⁰	
19. hr	Ir ¹⁹⁴	years	Rb ⁸⁷
19.3 hr	P ¹⁴²		

precision resistors. This is used to monitor the power level of the WBNS. Absorbing samples are placed in a specially fabricated graphite sample holder which is inserted in a reproducible geometry in the central exposure facility. The control-rod position is indicated on a 36-in. steel scale which is mounted alongside the channel in which the outer extremity of the control-rod assembly moves external to the shield. A vernier fixed to the end of the control rod slides along the steel scale. This arrangement permits one to determine the position of the rod to ± 0.001 in.

Table 1-10 Comparison of Radioisotopes Produced by 1-watt WBNS

Half-life	Number of radioisotopes	
	Produced by WBNS at 1 watt	Available for distribution from ORNL
10 min-12 hr..	42	0
12 hr-4 days...	32	32
Over 5 days....	30	70

The control rod has been calibrated with standard samples containing 3.72 mg of boron in the form of Bakelite impregnated with boric acid. As would be expected from the location and configuration of the control rod, the reactivity control is a nonlinear function of the control position. At a rod setting of approximately 26 cm (the control rod withdrawn 26 cm from the reflector), the reactivity control has been determined to be equivalent to 5.91×10^{-4} cm² of absorption cross section per mil of control rod. This is based on a neutron-absorption cross section for boron of 750 barns. In this region it is found that the critical position of the control rod can be determined to within ± 0.0002 in., so that the uncertainty in cross section for a given critical determination is $\pm 1.2 \times 10^{-3}$ cm². Since two determinations of the critical position are required for a cross-section measurement, the uncertainty resulting from the control-rod settings alone is then only $\pm 1.7 \times 10^{-3}$ cm².

From period measurements, the reactivity control of the rod in the afore-mentioned region is

determined to be 0.0141 inh/mil of rod. Using the inhour equation for a water-boiler-type reactor, this is found to correspond to 4.33×10^{-5} per cent reactivity per mil of rod.

At a control-rod setting of about 60 cm, the effect of the control rod is equivalent to 8.3×10^{-5} cm² of absorption per mil of rod. In this region the critical position of the rod can be determined to within ± 0.008 in., so that the uncertainty in a cross-section measurement resulting from the control-rod settings is $\pm 10^{-3}$ cm².

The great advantage in sensitivity of the WBNS for measurements using this technique is shown when one determines the effect on reactivity for a given cross section from the above data. This constant for samples placed in the central exposure facility of the WBNS is found to be 4.19×10^{-2} cm²/inh. This can be compared to the constant obtained for the Argonne graphite reactor (CP-2) of 2.05 cm²/inh.

One of the problems associated with the danger-coefficient technique is that of maintaining minimum extraneous reactivity changes in the reactor during the period required for the measurements. A method of surmounting this difficulty is to operate the WBNS at a very low power level, approximately 0.2 watt, so that changes resulting from temperature effects, power coefficients, and other variations in reactivity associated with high-power operation are minimized. The effects of any reactivity drifts are also greatly decreased by performing the measurements in a cyclic time sequence; that is, the critical position of the control rod is obtained for sample A, for sample B, and then for sample A again. The slow variation in reactivity is then averaged out.

To check the effects of possible variations in reactivity, a series of critical-position determinations has been made at a power level of 0.2 watt during which nothing was changed in the WBNS except for small changes in the control rod required to determine the exact critical position. During the period required for the data taking, the reactor reactivity was observed to decrease; that is, the control rod had to be withdrawn slightly to keep the reactor critical. Plotting the critical position as a function of time during the experimental run, it was determined that the reactivity changed at an approximate rate of 1.05×10^{-3} inh/min. It should be noted that the rate

of change of reactivity varies from time to time and is not always negative. The value given here appears to be typical. A similar set of data was then taken at an operating power level about three times greater than the one used above. Again the reactivity was observed to decrease, but at a rate of only 1.77×10^{-3} inh/min. Phenomena that are strongly suspected of contributing to this slow variation of reactivity with time include a slight heating of the solution by the absorption of the fission energy, the possible build-up of the gases of dissociation in the solution, or a combination of these along with others.

In connection with the above problem, it is of interest to examine the uncertainty that a drift in reactivity of the magnitude observed would introduce in a cross-section measurement. The time difference between the two critical-position determinations necessary for a cross-section measurement is about 1 hr. During this time a reactivity drift of approximately 6.3×10^{-2} inh might be expected on the basis of the above data. Thus, if no correction for the drift is made, an uncertainty in the cross-section determination of only about 2.6×10^{-3} cm² might be introduced. However, this uncertainty can be reduced considerably by making the cyclic measurements mentioned above.

It may be stated that the WBNS has proved to be an extremely versatile and useful tool for a great many varied research programs. This type of reactor gives a very high neutron flux per unit power over a small region. The use of this particular design of a water-boiler-type reactor is somewhat limited because of its low power rating. However, with suitable design changes, the power rating can be increased manifold so that a neutron flux of 10^{12} neutrons/cm²/sec is easily obtainable.

FIVE-HUNDRED-WATT SOLUTION-TYPE REACTOR

General Design Features. The 500-watt solution-type reactor consists of a spherical-core tank slightly more than 1 ft in diameter, which contains uranyl sulfate fuel solution and is embedded in the center of a right-circular cylinder of graphite 5 ft in diameter and 5 ft high. The graphite, which serves as a reflector, is contained in a

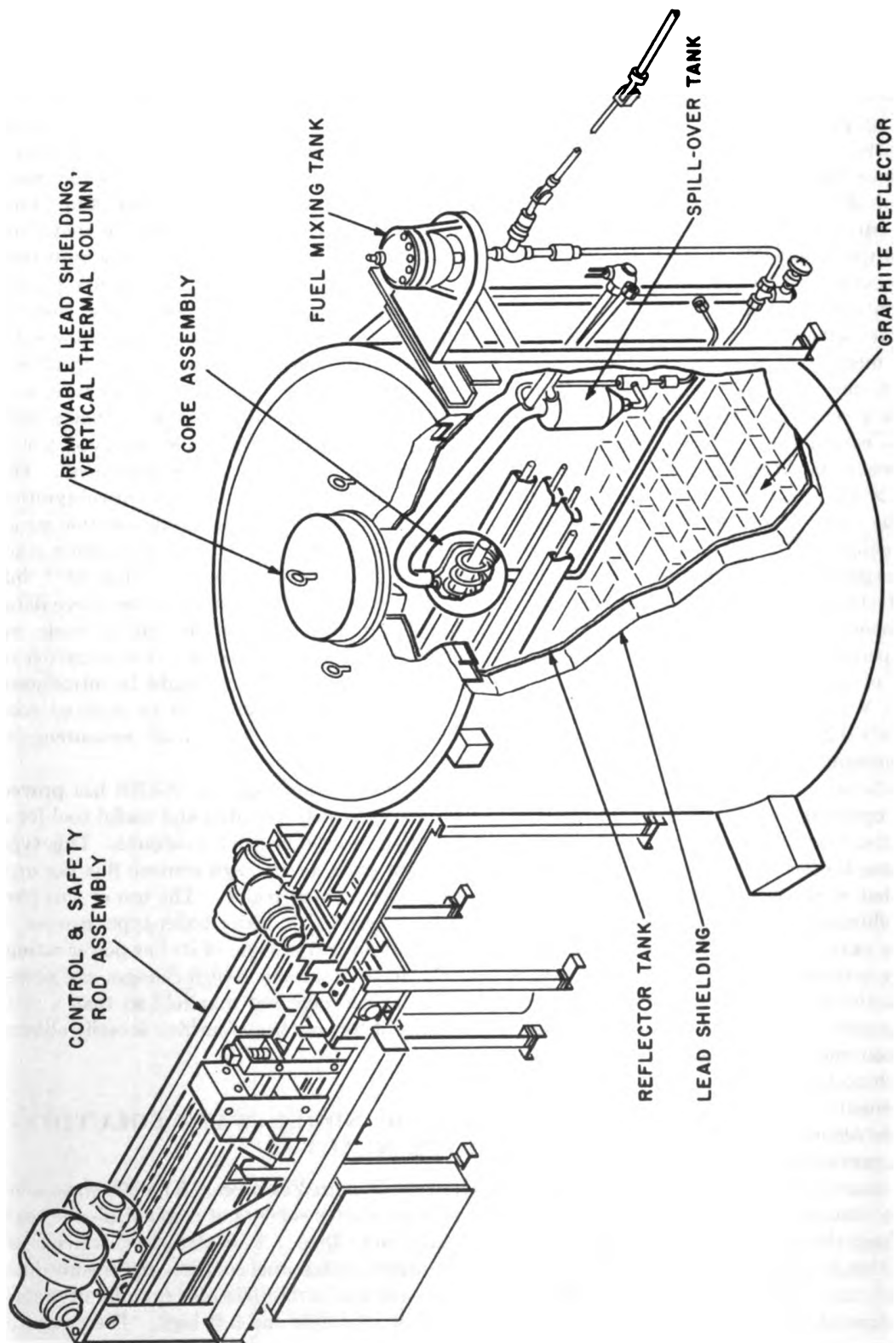


Fig. 1-33 500-watt solution-type reactor, isometric.

cylindrical steel tank. A 5-in.-thick layer of lead on the reflector tank provides shielding against radiation emitted by the core. The shielding is adequate to permit access to the inside of the reactor cave when the reactor is shut down.

The reactor is situated in a cave constructed from keyed 3-ft-thick concrete blocks. The out-

and terminates at the outer edge of the shielding, allowing access to the high-flux region in the center of the sphere. The experimental facilities are further complemented by eight removable stringers on the south side of the reactor which extend to the farthest side of the sphere on two sides. The stringer bars measure 4 by 4 in. in

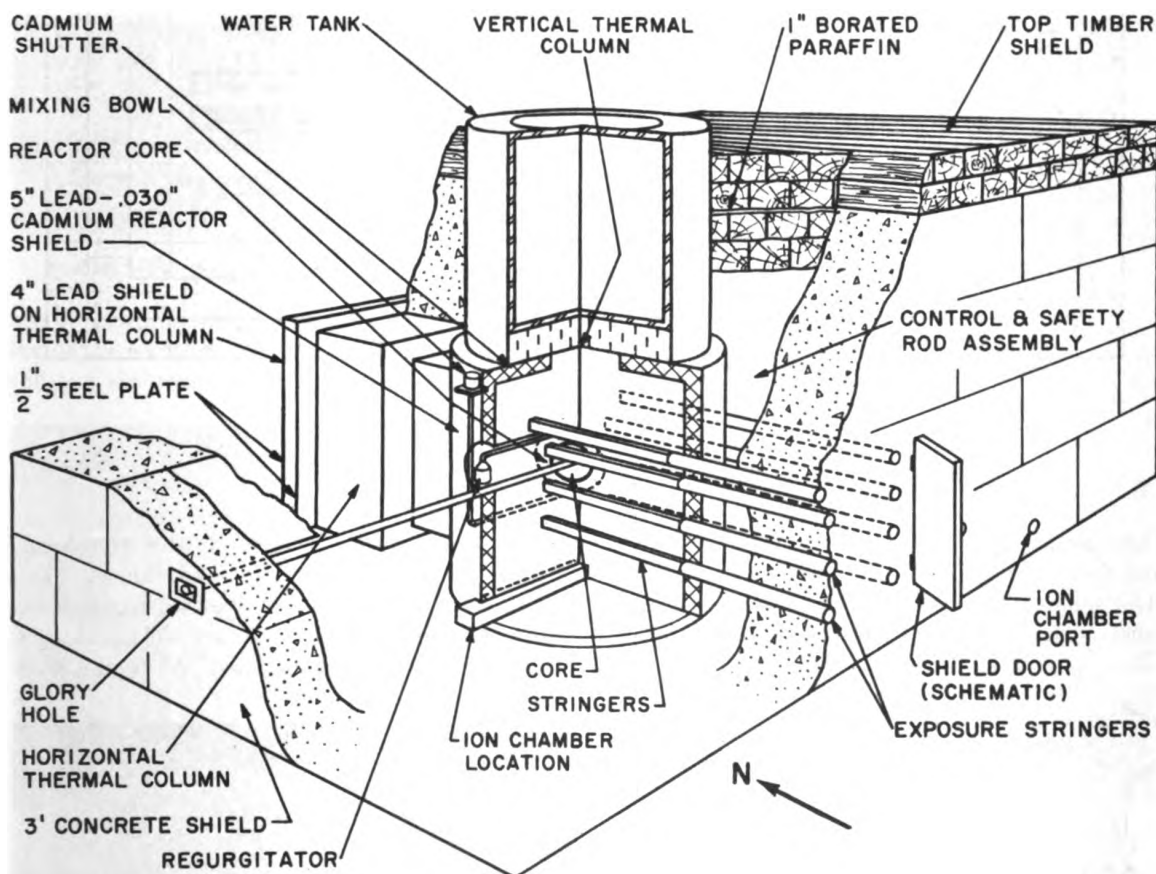


Fig. 1-34 Reactor installation, isometric.

side dimensions of the cave are 13 by 26 by 9 ft high. The top of the cave is covered by 2 ft of timber, beneath which is 1 in. of borated paraffin and 2 ft of timber recessed into the cave. Removable timber hatches allow access to the interior of the cave.

A maximum flux of 2×10^{10} thermal neutrons/cm²/sec is provided at the center of the spherical core when the reactor is operated at the design power. A 1.1-in.-ID tube passes completely through the reactor and the center of the sphere

cross section by 3 ft long. The graphite and lead have been so arranged that a section 16 in. in diameter may be removed directly above the sphere. A flanged tank area 3 ft square was provided on the north side of the graphite for the addition of a horizontal thermal column.

Figure 1-34 shows a perspective cutaway of the reactor and cave. Figure 1-35 presents a top view of the reactor with the timber shielding removed, showing the relation of the various components of the system to the reactor.

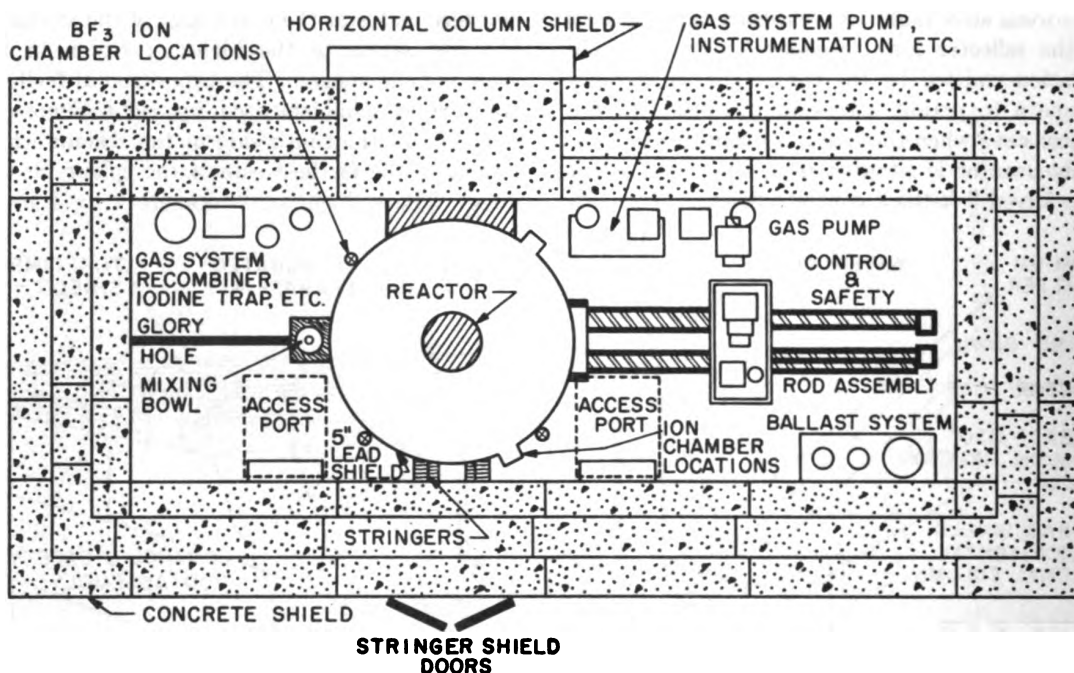


Fig. 1-35 Reactor installation, plan view.

Core and Fuel-handling System. The core and fuel-handling system consists of a stainless-steel sphere $12\frac{1}{2}$ in. OD, a stainless-steel mixing bowl located on the outside of the graphite reflector,

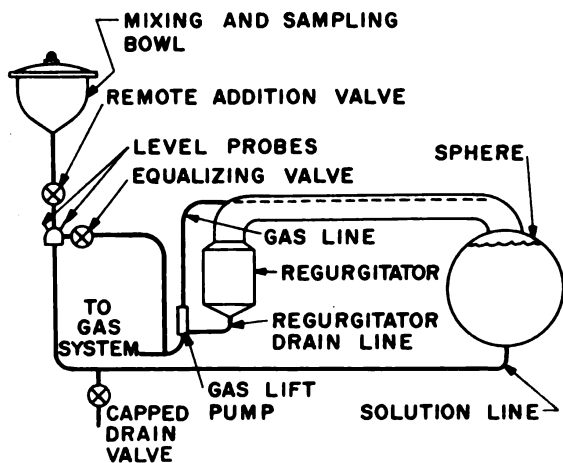


Fig. 1-36 Fuel system.

tor, a stainless-steel cooling coil inside the sphere, and a spillover tank directly connected to the spherical core. A flow sheet for the fuel-handling system is shown in Fig. 1-36.

The sphere is constructed from two spun hemispheres of type 316 stainless-steel sheet. The minimum wall thickness of the two hemispheres is 0.063 in. The hemispheres are welded along a circumference in a plane tilted 15° to the horizontal.

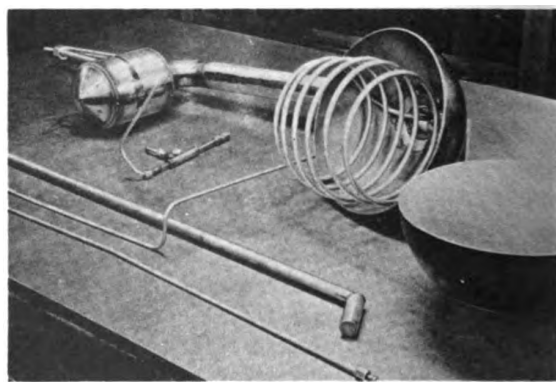


Fig. 1-37 Core and regurgitator during assembly.

A type 316 stainless-steel tube $1\frac{1}{4}$ in. OD with a $\frac{1}{16}$ -in. wall is welded through the sphere along the horizontal diameter of the sphere, forming the central exposure facility, in the reactor core.

A stainless-steel tube $\frac{3}{16}$ in. ID connects the bottom of the sphere to the mixing bowl through a stainless-steel gate valve. A valve and T joint

tubing $\frac{5}{16}$ in. ID by 160 in. long. The coolant lines and gas-handling lines all enter the sphere at the top. The connections to the entrance and

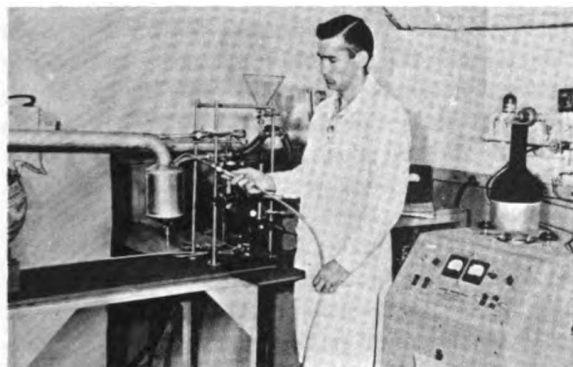


Fig. 1-38 Core and regurgitator assembly.

are provided in this line to permit gravity draining of the solution from the sphere. The cooling coil is a six-turn helix of type 316 stainless-steel

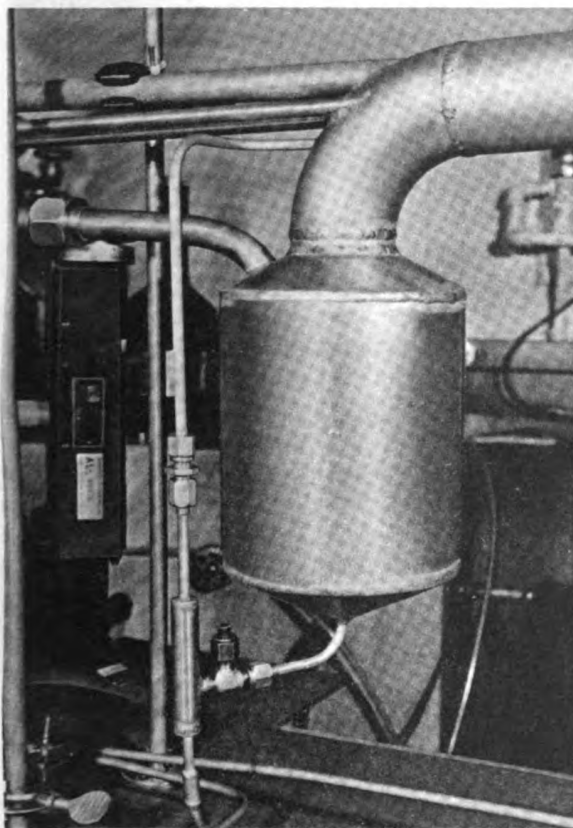


Fig. 1-39 Regurgitator and lift pump with associated piping.

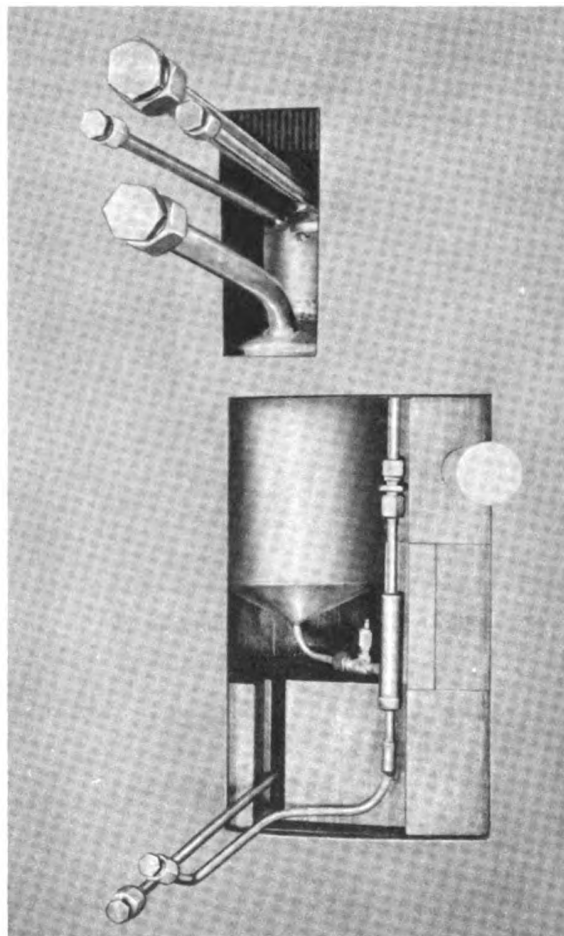


Fig. 1-40 Reflector tank, showing piping to core after installation in the reflector.

exit lines of the cooling coil, the gas-handling system, and the fuel mixing bowl are all made with compression-type fittings located just outside the graphite reflector (see Fig. 1-37).

The spillover tank is located inside the graphite reflector about 27 in. from the center of sphere. This tank, into which the fuel solution is expelled in the event of a nuclear "runaway" in the reactor, has a volume of approximately 2 liters. Any fuel collected in the spillover tank is returned slowly to the core by means of a gas lift pump, which operates continuously during operation of the gas-handling system. The completed core

assembly and the individual components are shown in Figs. 1-37 to 1-39. A view of the connecting lines to the core, after installation in the reflector tank, is shown in Fig. 1-40.

The mixing bowl is fabricated of type 316 stainless steel and has a volume of about 3 liters. Fuel to be added to the core is first put into this mixing bowl and then allowed to drain by gravity

into the core by opening the valve directly under the bowl. A level indicator, consisting of two electrical probes designated as high and low, indicates the fuel level in the line to the bottom of the mixing bowl. When either of the probes makes contact with the fuel solution, a neon lamp on the instrument panel lights. In order to check the liquid level in the sphere, the blower must be shut off and the equalizing valve must be opened to equalize the pressure at the indicator with that in the sphere.

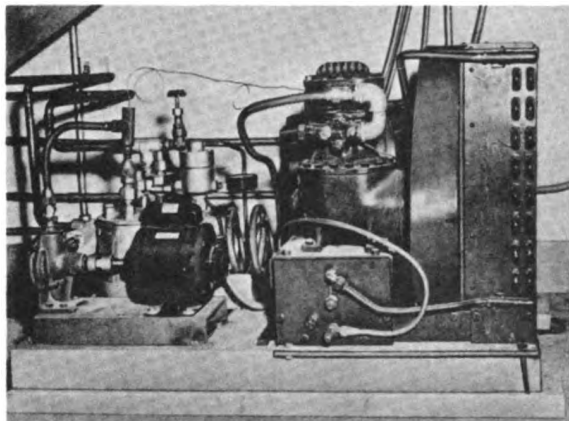


Fig. 1-41 Cooling-system installation.

Cooling System. The cooling system consists of a Freon refrigeration system, heat exchanger, coolant-circulating pump, core cooling coil, cooling coils for the condensers in the gas-handling system, and suitable controls for maintaining proper temperatures of the coolant. The system installation is shown in Fig. 1-41; a flow sheet for the system is shown in Fig. 1-42.

The refrigeration system is a Freon condensing system, with a rating in excess of 12,000 Btu/hr. The Freon compressor is operated by a 1-hp 220-volt three-phase electric motor. A pressure-

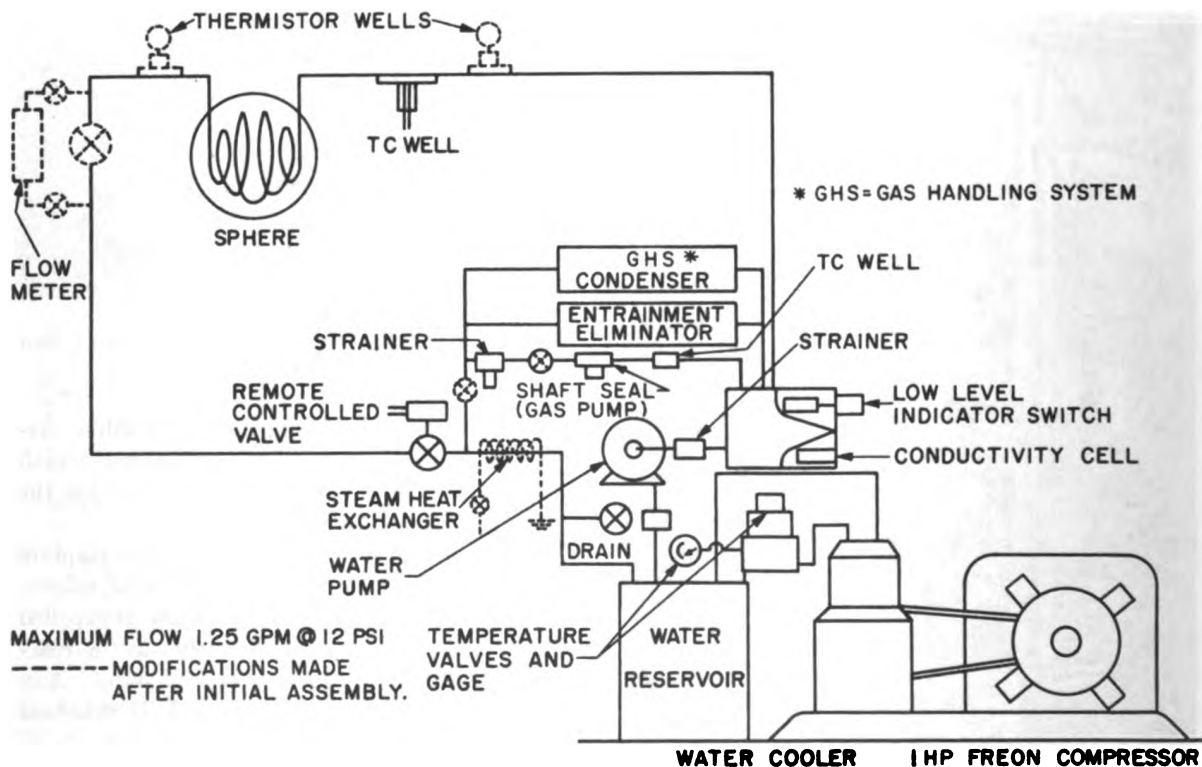


Fig. 1-42 Cooling system, schematic.

actuated switch controls the operation of the motor to give the proper flow of Freon through the heat exchanger.

Located in the Freon line into the exchanger is a pressure-regulator valve which can be manually adjusted to control the temperature of the Freon in the exchanger. A pressure gauge located adjacent to the pressure regulator valve is calibrated

in the sump. The resistivity of this water is maintained at 100,000 ohms-cm. A level indicator on the sump indicates a leak by a light on the control panel when the level is low.

Gas-handling System. The gas-handling system of the reactor is a closed system in which the reactor atmosphere gas is circulated by pumping,

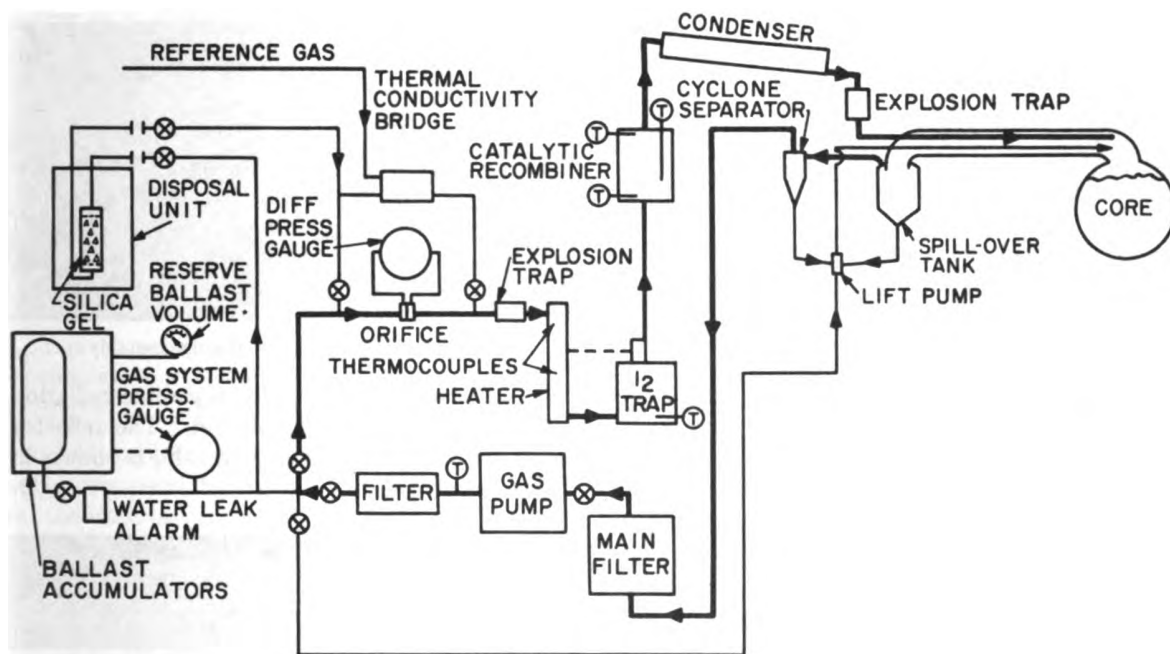


Fig. 1-43 Sweep-gas system, schematic.

to read the Freon temperature in degrees Fahrenheit. To reduce cycling of the Freon-pressure regulator, an additional cutoff switch has been installed in the condenser unit in series with the pressure switch. This second switch is actuated by a temperature-sensitive bulb installed in a well in the coolant-water line.

The coolant, which is distilled water, is circulated through the system by a 2.2-gpm centrifugal pump. The rate of coolant flow in the core cooling coil is adjusted by a remote-controlled motor-driven throttle valve located in the coolant line. The position of the valve is indicated on the control panel so that the necessary amount of coolant can be circulated to maintain a given temperature for any given power level. The electrical conductivity of the water is indicated on the control panel from a conductivity cell located

at a rate of 3 to 5 cfm, over the fuel solution in the core tank and through the components of the system. The carrier gas picks up water vapor, volatile fission products, and dissociated hydrogen and oxygen from the space above the fuel solution. The hydrogen and oxygen are catalytically recombined, and the resultant water vapor, together with the vapor initially present in the stream, is condensed and returned to the core tank. In addition to the recombination feature, the gas-handling system provides for the storage of volatile fission products, for the maintenance of the total pressure in the closed system over a range of values below atmospheric pressure, and for the disposal of radioactive gases whenever desired.

The functions of the components of the gas-handling system may be understood from the schematic-flow diagram in Fig. 1-43. The reac-

tor gas atmosphere is swept, in sequence, through cyclone separator, filter, gas pump, second filter, explosion trap, heater, chemical trap for removing iodine, catalytic recombiner, condenser, second explosion trap, and then back to the core tank. Downstream from the second filter, a side stream of the main gas flow is bypassed through a nozzle which is arranged to act as a lift pump. This lift pump returns to the core tank, at a slow rate, any liquid which drains from the spillover tank. Pressure in the reactor atmosphere system is regulated by ballast accumulators, in which the volume of the gas storage space is regulated by water pressure applied to the flexible walls of the space. Rupture of the flexible walls actuates a warning light on the control console. Operation of the ballast system permits maintenance of gas-system pressure from 2 to 6 in. of water below atmospheric. The ballast system is actuated automatically by preset limit switches on the gas-system pressure gauge. Ballast capacity is indicated on the control console and serves as an indication of a leak in the gas system. Total pressure is measured by one gauge, and flow rate is determined by means of a second gauge, connected across an orifice. Both pressure and flow rate are indicated on the control console. Also indicated are reactor gas temperatures measured at the gas-pump exit, the chemical iodine trap, and the hydrogen-oxygen catalytic recombiner. The hydrogen concentration is determined by a thermal conductivity bridge, one of whose arms is immersed in the original reactor atmosphere gas as reference. The percentage of hydrogen in the system is indicated on the control console.

The activity of the reactor atmosphere gas can be reduced considerably by the use of a disposal unit in which krypton and xenon are absorbed on silica gel. Alternatively, the reactor atmosphere gas may be collected by valving at intervals into a shielded, previously evacuated, steel tank. The gas removed may be replaced by addition of gas to the reactor system, if desired.

Graphite Reflector. The reflector is a right-circular cylinder of graphite. The graphite is contained in a cylindrical stainless-steel tank 54 $\frac{3}{4}$ in. high with an inside diameter of 60 in. The spherical core is located along the vertical axis of the graphite cylinder with the center of

the sphere 34 $\frac{1}{2}$ in. from the bottom of the steel tank. The steel tank is enclosed with cadmium sheet 0.030 in. thick for thermal-neutron shielding.

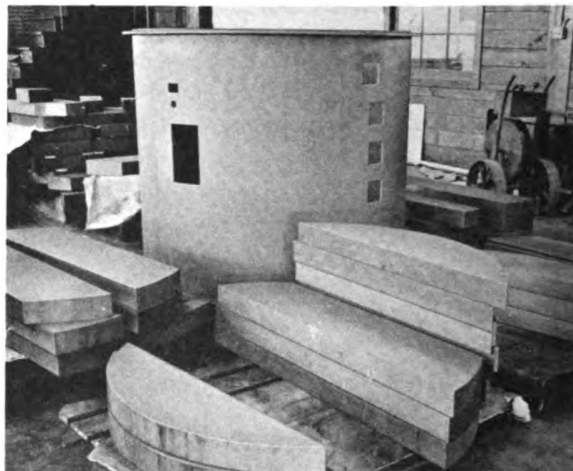


Fig. 1-44 Reflector tank during assembly.

This in turn is surrounded by a gamma-radiation shield consisting of 5 in. of lead. The reflector tank, in various stages of assembly, is shown in Figs. 1-44 to 1-46.

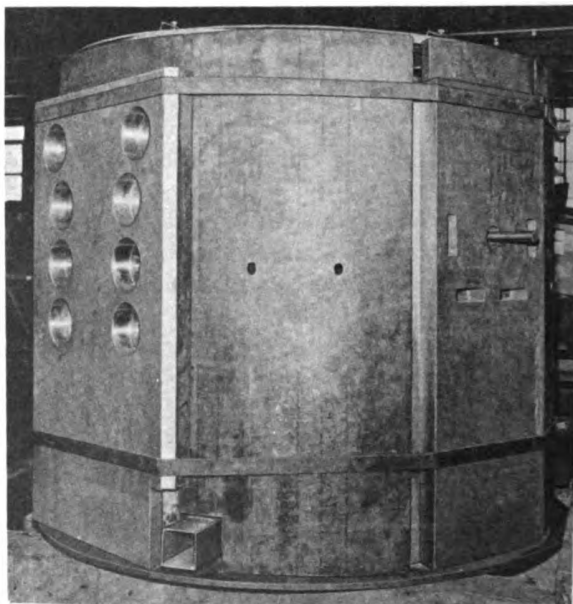


Fig. 1-45 Assembled reflector tank with lead shielding in place.

The graphite blocks are stacked inside the tank in two groups, designated the stringer graphite

and the core graphite. The graphite stringer assembly, which comprises the main body of the graphite, is stacked in 14 layers. Each layer has

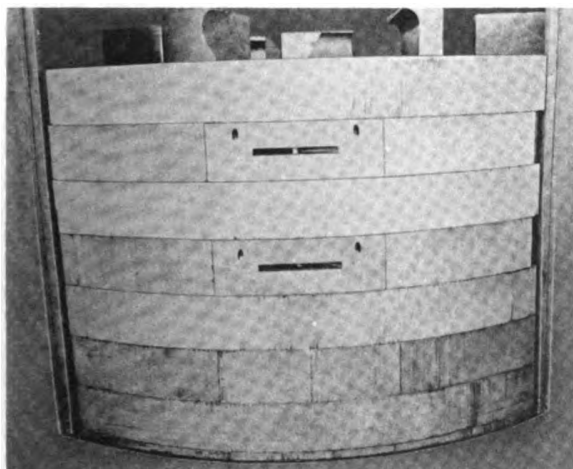


Fig. 1-46 Reflector tank during assembly with the flange assembly removed, showing provision for horizontal thermal column.

been laid in place and then secured with wooden wedges between the ends of the graphite blocks and the walls of the steel tank. A step-cut cavity is provided in the stringer graphite for the core tank and core graphite.

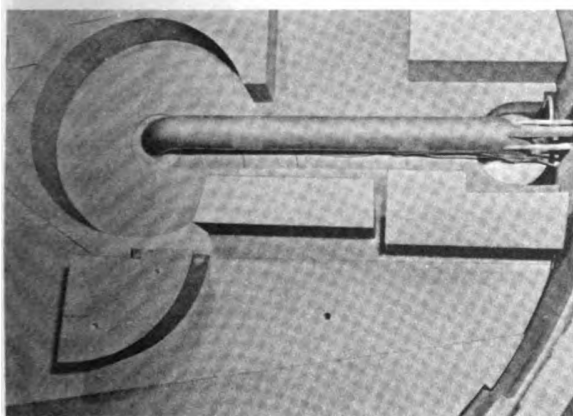


Fig. 1-47 Graphite reflector, from above, during assembly.

The core graphite extends through layers 7 to 13, inclusive. It fits somewhat loosely around the stainless-steel core assembly and can be easily removed if desired. The installation of the core is shown in Fig. 1-47. Those blocks of graphite

which might shift into openings around the core assembly are doweled in place with aluminum dowel pins $\frac{3}{8}$ in. in diameter and 7 in. long, each being provided with a $\frac{1}{4}$ in.-20 tapped hole to permit the insertion of a pull screw for easy removal. In each layer of the core graphite, there is one block which has been provided with lift holes which are tapped for a $\frac{3}{8}$ in.-16 screw.

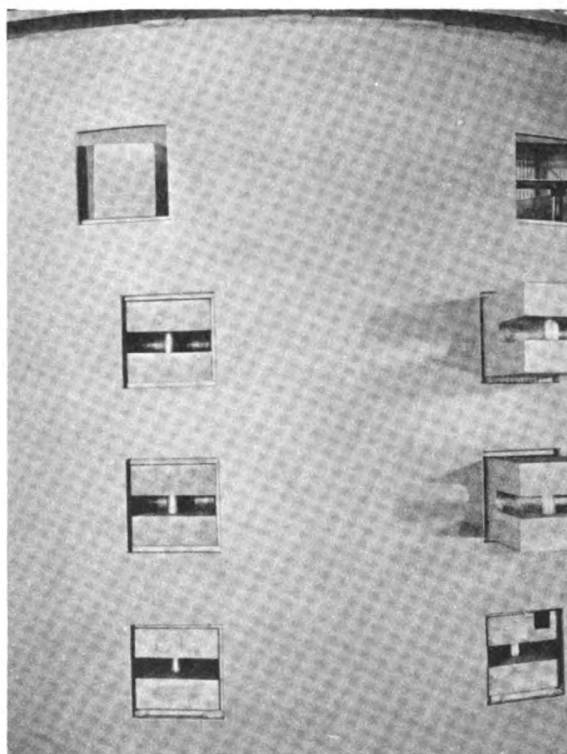


Fig. 1-48 Reflector tank, showing removable graphite stringers.

The graphite blocks directly above the spherical core tank form a removable vertical thermal column 16 in. in diameter by 18 in. high immediately over the reactor core. A separate shielding cover for this graphite region is provided. One side of the reflector tank has been provided with a flanged area 3 by 3 ft. The gamma-shielding material is removable in this area, permitting the stacking of additional graphite outward from the main body of the reflector graphite to form a horizontal thermal column. Other experimental facilities in the core and reflector include a central exposure facility, which extends through the reflector and the spherical core, and eight remova-

ble graphite stringers located about the core, as shown in Fig. 1-48. The central-exposure-facility hole through the graphite is lined with a 3 S-O aluminum tube which is aligned with a stainless-steel tube through the spherical core. The eight removable graphite stringers are 4 in. by 4 in. by approximately 36 in. long and can be completely

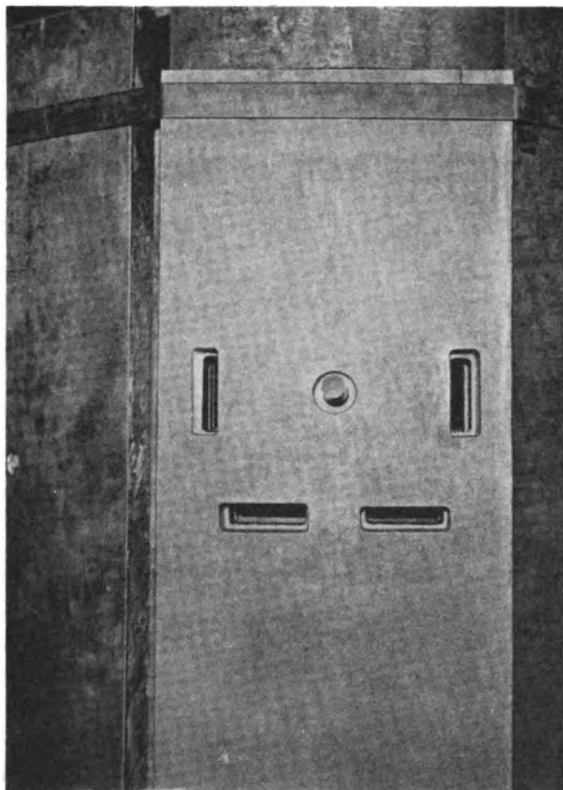


Fig. 1-49 Reflector tank with lead shielding in place around the control- and safety-rod slots.

removed from the reflector through the holes in the shielding. Four horizontal slots 4 by $\frac{3}{4}$ in. are provided through the reflector for the safety and control rods. These are shown in Fig. 1-49.

Control and Safety Rods. The reactivity control of the reactor is maintained with a control- and safety-rod system consisting of two safety rods and two control rods. The rods all enter the reflector in horizontal slots which are tangential to the sphere. The control-rod assembly being installed is shown in Fig. 1-50. Plan views of the control- and safety-rod system, along with

the locations and functions of the microswitches which control the operation of the rods, are shown in Fig. 1-51.

The two safety rods are identical in construction and are withdrawn from the reflector by electric-motor drive systems. The drive systems as mounted on the assembly are shown in Fig. 1-52. The motors operate carriages to which the rods are connected with magnetically actuated latches. The speed of withdrawal of each rod is such that the rod is completely removed from the reflector in approximately 7 sec. If at any time the magnetic latches are deenergized by any one of the scram circuits, the safety rods are accelerated into the reflector by weights suspended in the weight wells. The weights are attached to the rods with flexible cables which pass over pulleys. The rods are stopped at the end of their travel by friction brakes with adjustable spring-loaded brake shoes, as shown in Fig. 1-53.

The neutron-absorbing component of each safety rod consists of a type 321 stainless-steel sleeve $37\frac{1}{2}$ in. in length with a cross section of $3\frac{3}{4}$ by $\frac{1}{2}$ in., into which approximately 2.2 lb of boron carbide has been packed. The outside end of the absorbing component of each rod is fabricated from lead, which acts as a shielding plug when the rod is completely inserted in the reflector.

The two control rods are similar in construction to the safety rods but serve in quite different capacities. The coarse-control rod, shown in Fig. 1-54, controls approximately 1.4 per cent in reactivity and is used to bring the reactor up to power. The regulating rod (Fig. 1-54) controls about 0.75 per cent in reactivity and is equipped with a servodrive, operated by the servoamplifier. Thus it can be used as an automatic power regulator during normal reactor operation.

The coarse-control rod has a neutron-absorbing component identical to that in the safety rods. The electrical drive system is a $\frac{1}{8}$ -hp motor and gear box which is fed through a two-speed magnetic clutch to drive a pinion gear engaged in the rack on the coarse-control-rod extension. The magnetic clutch gives two speeds of movement to the coarse-control rod, 0.5 cm/sec and 0.05 cm/sec. In the event of a scram, the fast speed is automatically selected to drive the coarse rod into the reflector. Attached to the rod drive is a 50-

volt 50-cps selsyn transmitter, which furnishes a signal to a dual selsyn receiver on the instrument console for remote indication of the rod position.

and only about 1 lb of this material is contained in the assembly. The electrical drive system for the regulating rod is a control motor which feeds

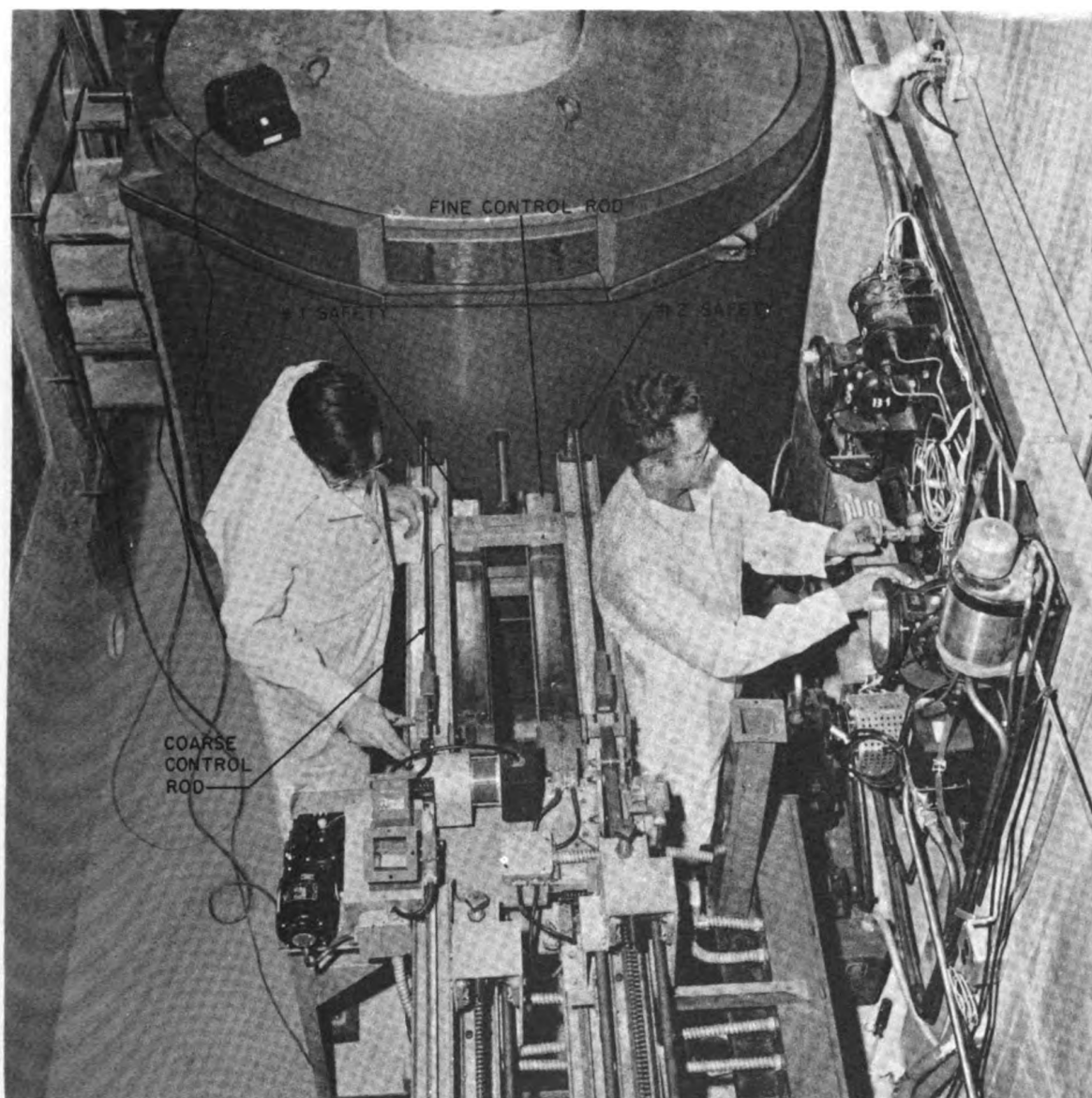


Fig. 1-50 Control- and safety-rod installation.

This position indicator gives the rod position to within 0.002 cm. The coarse-control-rod drive assembly is shown in Fig. 1-55.

The regulating rod is similar to the coarse-control and safety rods. However, the region in which the boron carbide is contained is smaller,

through a speed-reducer gear box, with a reduction ratio of 200:1, to a pinion gear, which engages the rack on the rod extension. This assembly moves the regulating rod at a maximum speed of 0.5 cm/sec. A remote-indication system gives the position of the rod.

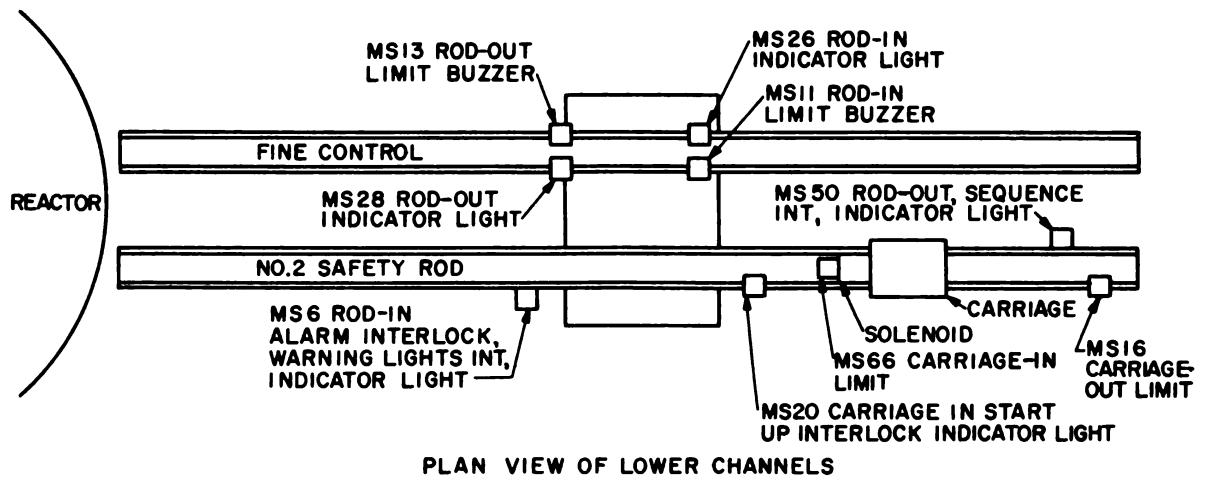
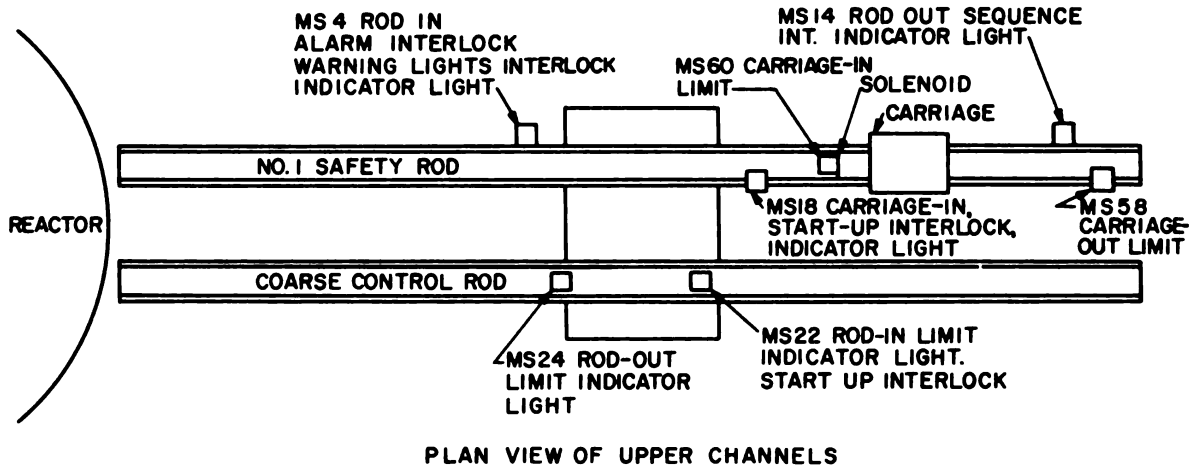


Fig. 1-51 Schematic, showing locations and functions of control- and safety-rod microswitches.

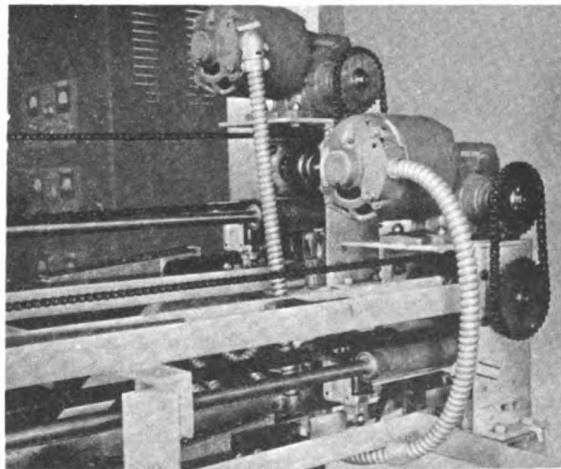


Fig. 1-52 Drive system for safety rods.

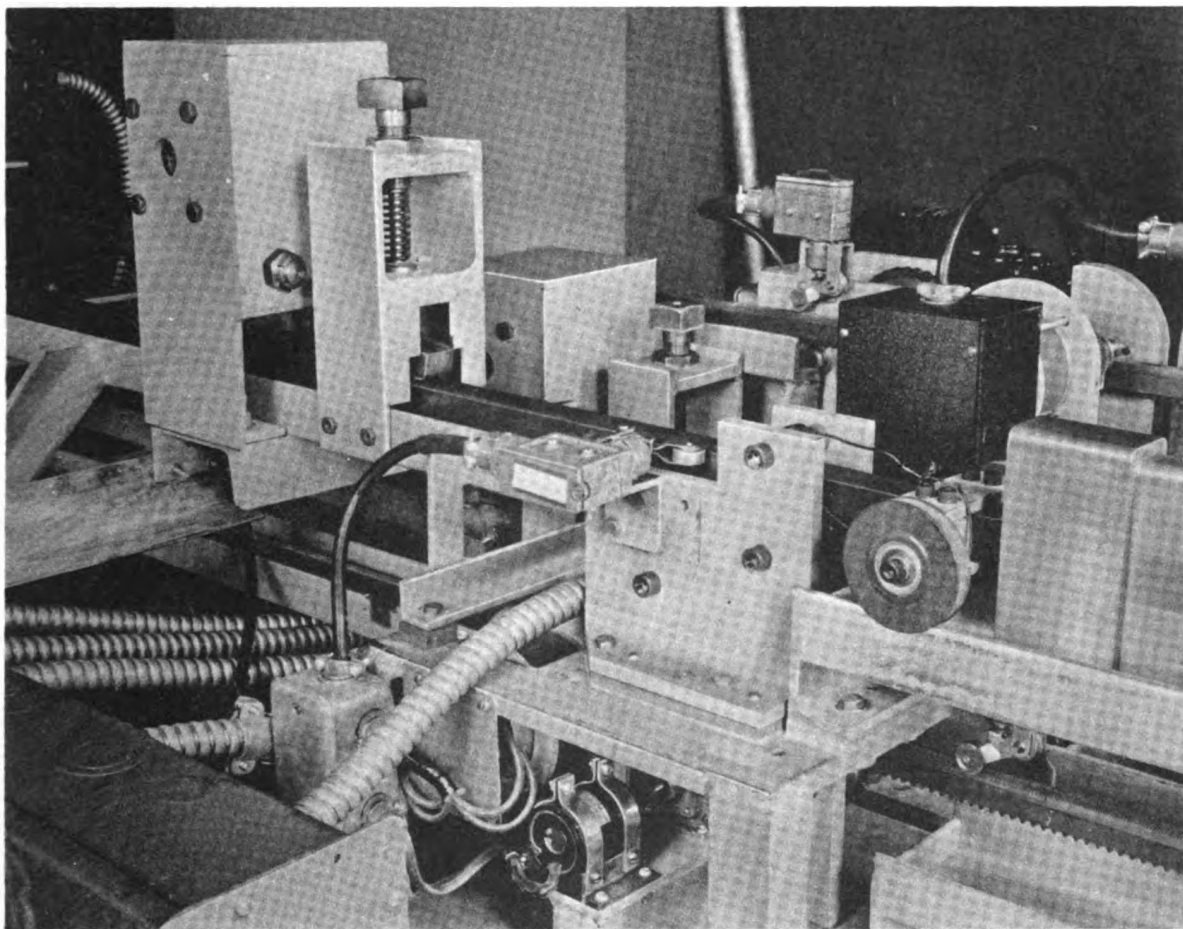


Fig. 1-53 View of control- and safety-rod assembly, showing brake system for safety rods.

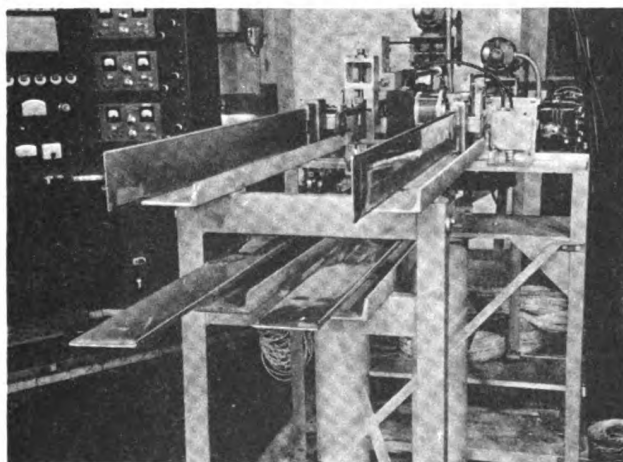


Fig. 1-54 Control- and safety-rod assembly prior to installation.

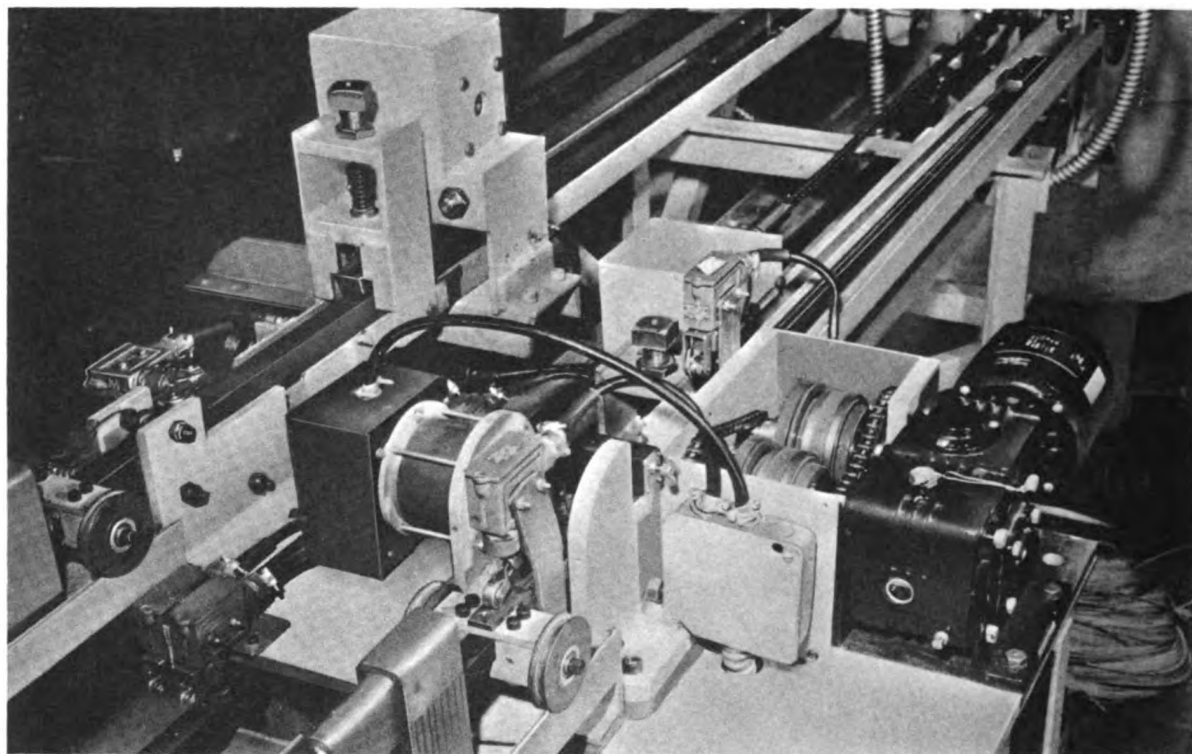


Fig. 1-55 View of control- and safety-rod assembly, showing drive system for coarse-control rod.

The motion of the regulating rod can be controlled manually or automatically with the servoamplifier. The servoamplifier receives its signal from one of the ionization chambers used to monitor the neutron level of the reactor.

Instrumentation. The reactor instruments consist of neutron and gamma detectors and devices for measuring temperatures, pressures, liquid levels, flow rates, conductivity, and rod position. All instruments are remote indicating and are read or recorded on the control panel. The control console consists of five relay racks into which all of the recorders, indicating instruments, control switches, and relays are mounted. Figure 1-56 shows the console and control panel.

The neutron-detecting instruments are divided into two classes. Three enriched- BF_3 proportional counters are used from startup to 50 mw. They are located on the sides of the reactor outside the lead shielding. Their signals are fed into count-rate meters at the instrument panel. Three boron-lined gamma-compensated ionization chambers are located in channels in the base of the reac-

tor. These indicate power levels from 0.05 to 2000 watts. One of the chambers feeds a log N amplifier and recorder with associated power-level and period meters. The remaining chambers feed amplifiers which in turn drive power-level recorders. A selector switch allows connection of any of the BF_3 count-rate meters or one of two ionization chambers to a recorder. Scram signals are initiated by high-power indication on any of the detectors, and by a rapid power-level rise. Scram signals are also initiated by high pressure in the gas system, low gas-flow rate, high hydrogen concentration, or high or low core temperature. Figure 1-57 presents a block diagram of the scram signal system. Eleven temperatures are recorded on a 12-point recorder on the instrument panel. The temperatures recorded are the core (three locations), cooling water, cave, gas-pump shaft-seal cooling water, recombiner (three locations), gas pump, and iodine trap.

An earthquake detector mounted in the cave is included in the interlock system.

One gamma-ray ionization chamber is located on each side of the reactor in the cave. These

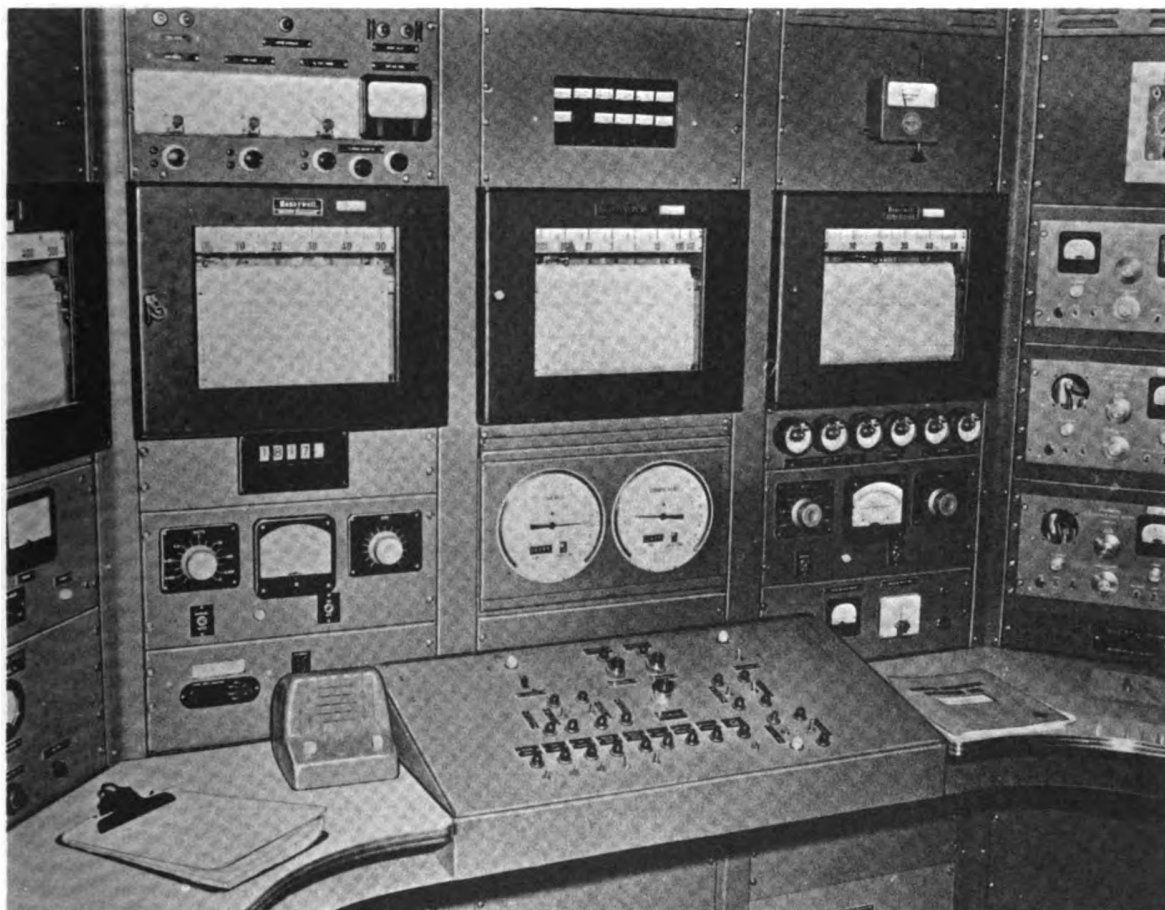


Fig. 1-56 Control-console installation.

indicate radiation levels up to 2000 roentgens per hour (r/hr) on the control panel. At 500-watt equilibrium operation, the maximum cave level is approximately 2000 r/hr.

Experimental Facilities. The principal facilities of the solution-type reactor for making neutrons and gamma rays available for research purposes are the central exposure facility, the removable graphite stringers, and the thermal columns. An evaluation of these facilities is given in the following paragraphs.

Central exposure facility. The central exposure facility, or glory hole, is a $1\frac{1}{8}$ -in.-ID hole which extends through the graphite reflector and the spherical core tank. It runs along a horizontal diameter of the reflector through the center of the core tank, as shown in Fig. 1-34.

Thermal-neutron distributions have been determined as a function of position within the reflector by indium- and gold-foil exposures in the glory hole. The distribution and indium-cadmium ratios are presented graphically in Figs. 1-58 and 1-59, respectively. A slight asymmetry in the distribution is caused by the presence of the spill-over tank, which presents a void near the west end of the glory hole. The cadmium ratio may be seen to dip rather sharply as the reflector edge and cadmium layer are approached.

Higher cadmium ratios and fluxes near and inside the core are possible if provisions are made for installation of reasonable quantities of materials having good slowing-down and low-absorption properties, for example, D_2O .

The gamma-ray intensity in the center of the sphere is approximately 1.5×10^5 r/hr.

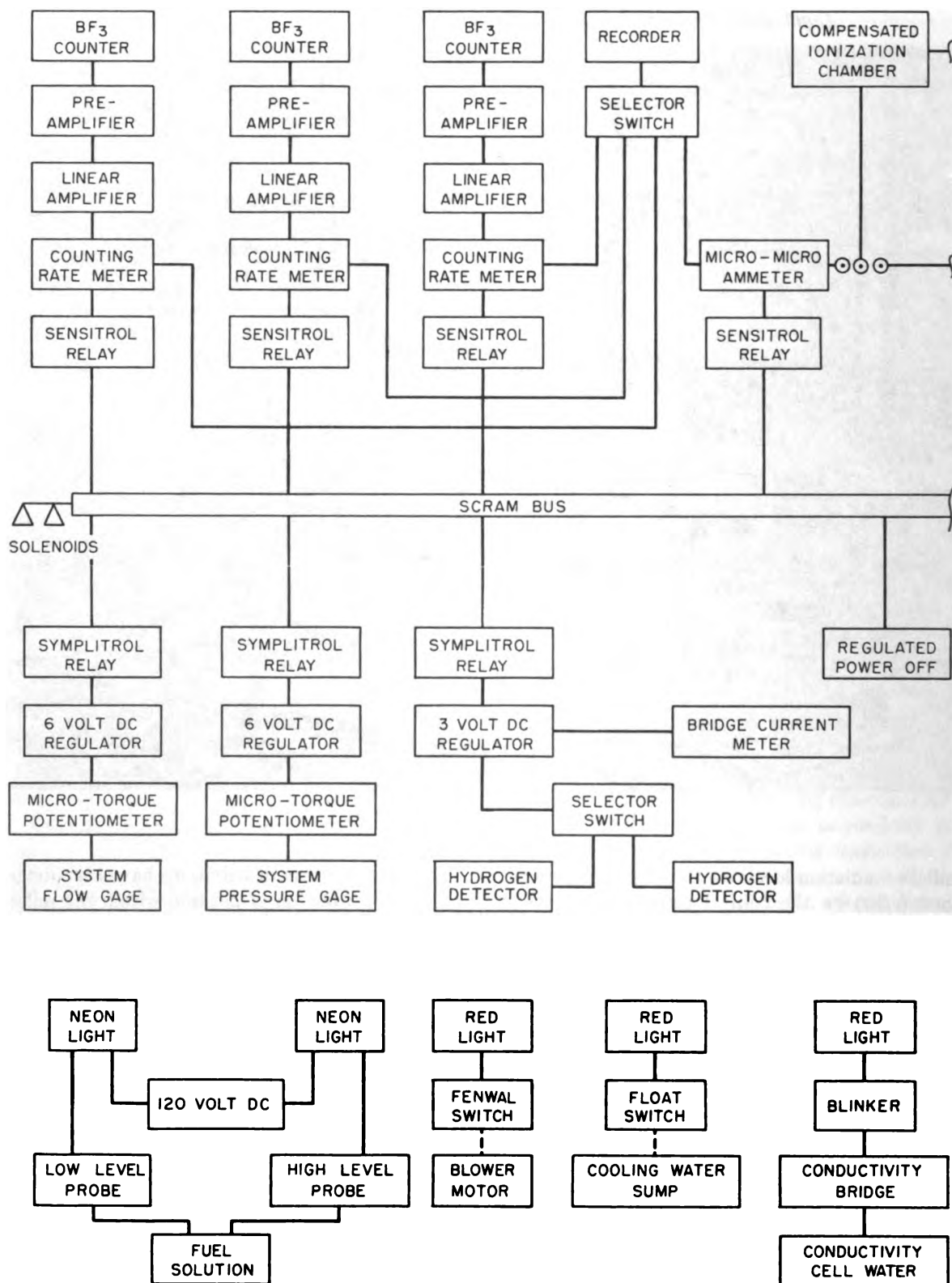
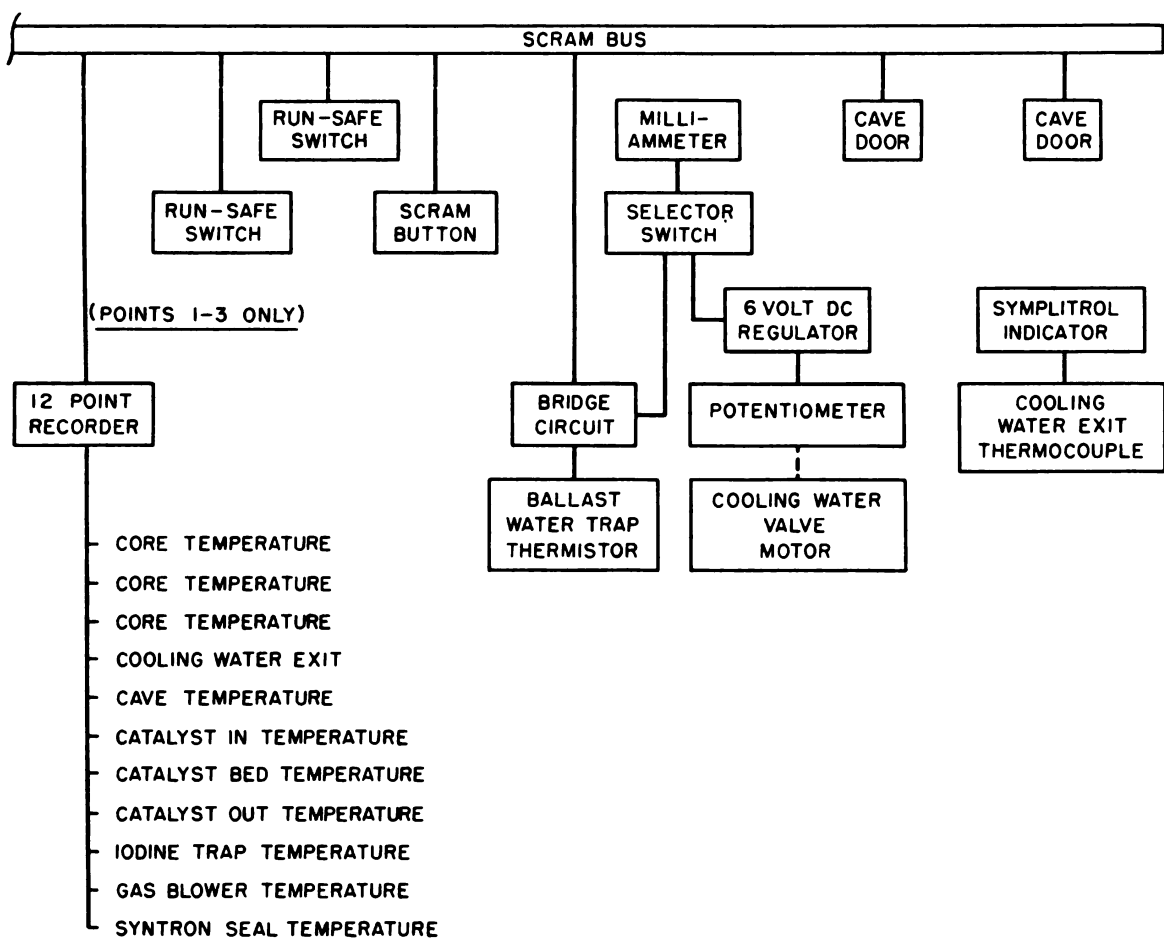
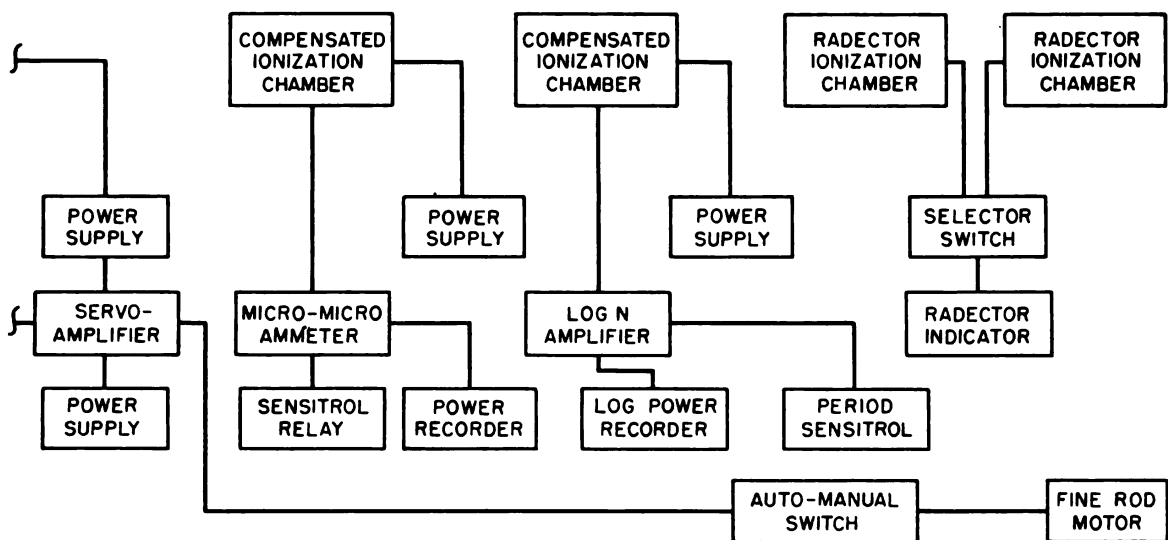


Fig. 1-57



Scram system.

Graphite stringers. Eight removable graphite stringers 4 in. square and approximately 36 in. long extend through the graphite reflector to the far side of the spherical core. They pass close to the core and are intended to carry samples into the reflector for exposure purposes. The stringers can be seen in Fig. 1-34.

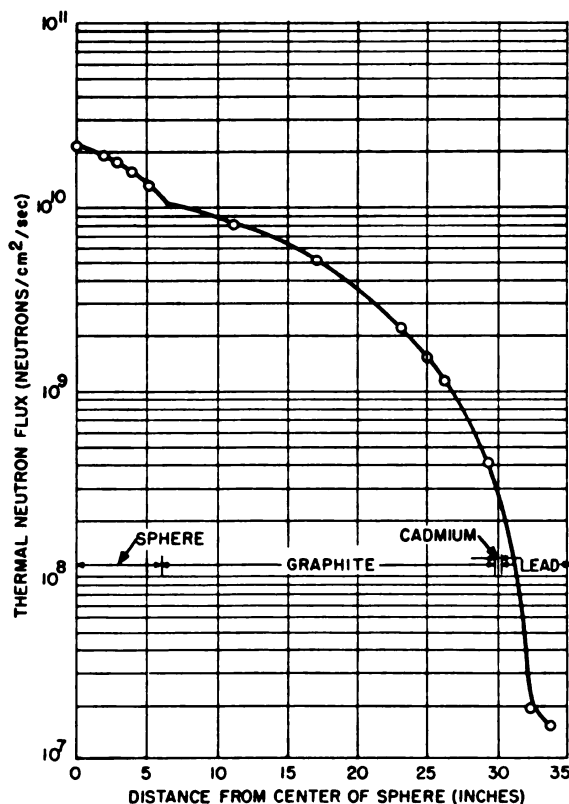


Fig. 1-58 Slow-neutron flux in glory hole.

Figure 1-60 presents the thermal-neutron distribution, as measured with indium foils, in the seven stringers normally used for long-term sample exposures. Also shown on the figure are the geometric relationships of those voids and reactor components which have measurable effects on the neutron distributions. In general, flux depressions are caused by voids, foreign materials, or the presence of control rods. For example, the saddle in stringer 2 traverse is caused by the presence of the spillover-tank-sphere pipe which passes just above the stringer and at approximately a right angle. Stringer 7 traverse shows a displacement and abbreviation of the peak due to the

fine-control rod used for control during the measurements. Stringer 3 displays a similar, although not as marked, displacement. The latter is caused by the presence of the No. 2 safety-rod void, which is in the vicinity. Ideally, stringers 3 and 7 should have identical traverses since they are mirrored images.

The foil exposures were made with the mock stringer bar shown in Fig. 1-61. The foils, $\frac{1}{2}$ -in.

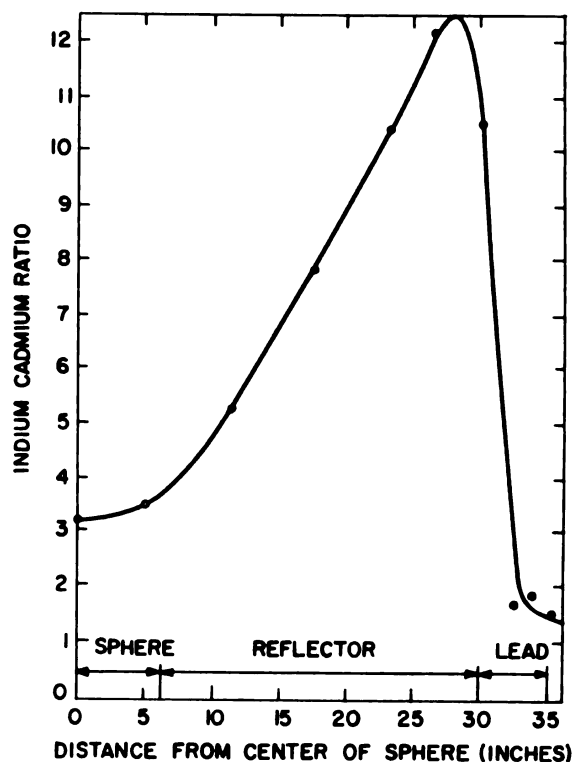


Fig. 1-59 Cadmium ratio in reflector glory hole.

disks, were positioned so that their centers were $1\frac{1}{2}$ in. above the major axis of the stringer bar. This was done because the actual stringers have rectangular recesses, of various widths, on the upper surface of the bars. Graphite filler pieces provide complete tamping and positive positioning for samples placed in the recesses.

In general, it is concluded that thorough exploration of exposure facilities is necessary if it is required that integrated-neutron-flux values be well known. If high productivity is desired, the reflector perturbations should be kept to a minimum by exclusion of voids and foreign materials.

Vertical thermal column. The graphite blocks forming the reflector have been stacked inside the reflector tank so that a vertical thermal column 16 in. in diameter and 18 in. high is formed directly above the reactor core, as shown in Fig.

nance neutrons. The measured diffusion length of the thermal neutrons is in good agreement with other reported values of 2.88 cm. The gold-cadmium ratio 1½ in. from the bottom of the tank is 201 with the cadmium shutter open and 122

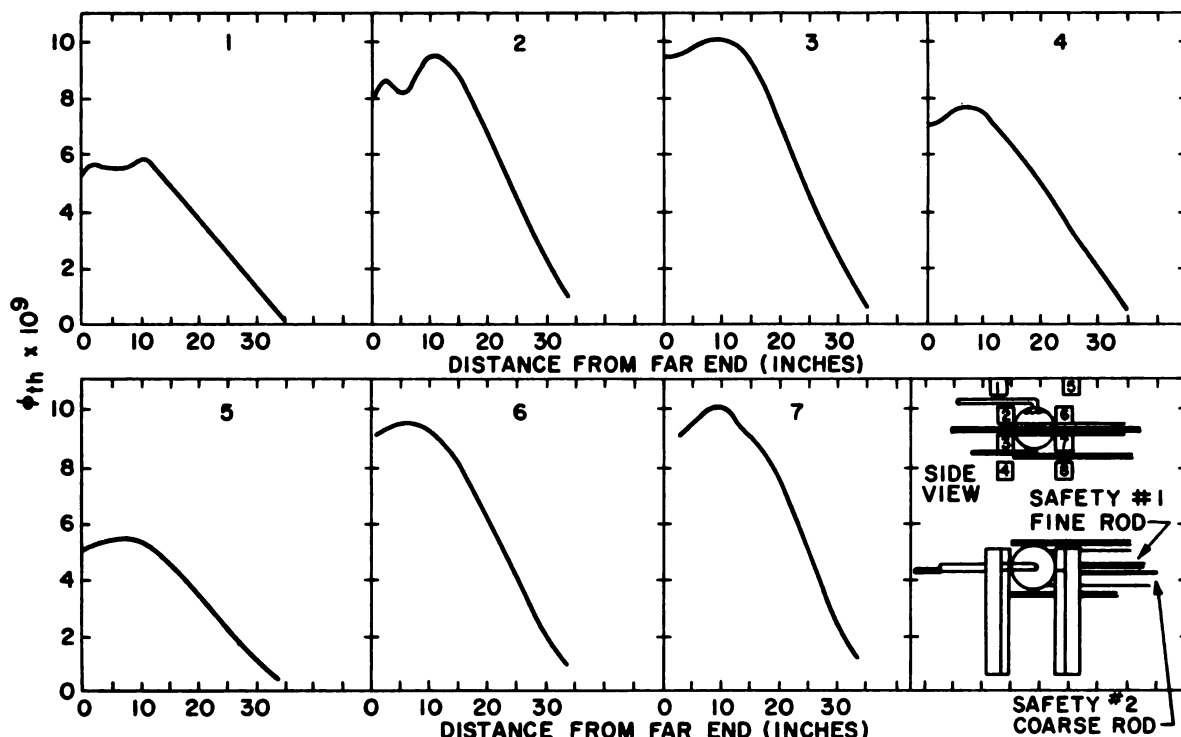


Fig. 1-60 Thermal-neutron distribution in stringers.

1-34. An aluminum tank 6½ ft in diameter and 5 ft deep, filled with light water, rests on top of the vertical column.

Neutron and gamma-ray distributions along the vertical center line in the water tank which

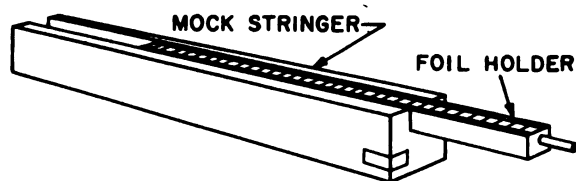


Fig. 1-61 Mock stringer bar used for foil exposures.

rests atop the vertical thermal column appear in Figs. 1-62 and 1-63, respectively. The brevity of the neutron information was necessitated by the rapid attenuation of thermal and indium reso-

with the shutter closed. Thermal fluxes measured at that point with the shutter open and closed are 7.8×10^6 and 5.0×10^6 neutrons/cm²/sec, respectively. The gamma-ray measurements were obtained with an ionization chamber which is believed to have an essentially flat energy response from 80 kv. The measurements were taken along the center line of the tank.

Although light water, because of its high absorption, is definitely not an ideal medium in which to make general neutron investigations, it does provide an extremely flexible facility. Experiments involving cavities and ducts or a wide range of sizes of biological specimens are particularly well suited to water-tank experiments. A water tank, equipped with proper inserts, has high potential for ease of beam production. The cost of such a facility is negligible in comparison to graphite thermal columns.

Horizontal thermal column. The horizontal thermal column is formed by stacking additional graphite outward from the main body of the re-

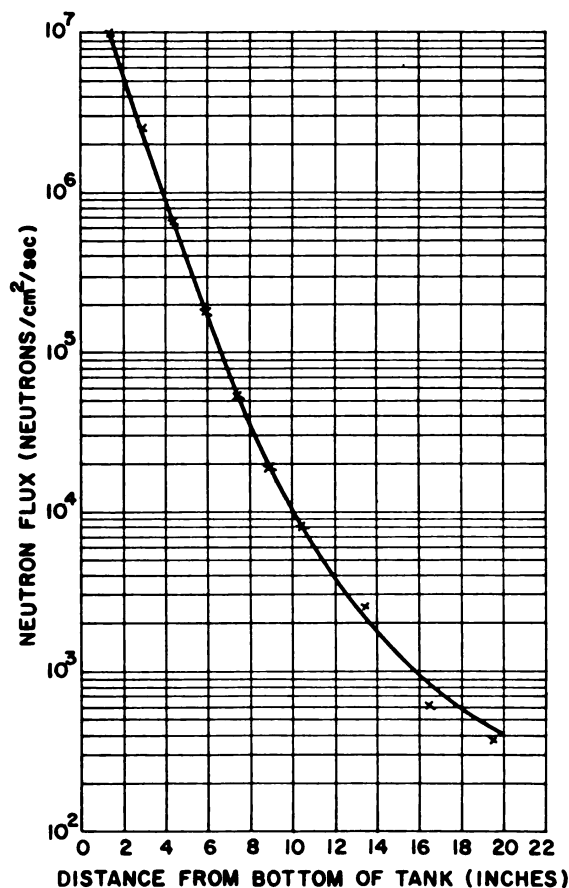


Fig. 1-62 Vertical neutron distribution in water-tank bare indium.

flector graphite, as shown in Fig. 1-34. The column is approximately 4 ft square and provides 6 ft of graphite between the reactor core and the outer face of the column. The column is reduced to 20 in. in cross section where it actually joins the reflector graphite. This was done to minimize the amount of lead removed for the juncture. This area is increased to 36 in. square at the outer surface of the lead, where a contour fit to the cylinder has been established. The cadmium was removed from the 36-in.-square area in the vicinity of the reflector-column interface to allow passage of neutrons. The horizontal-column working face is shielded for personnel protection by 1 in. of heavily borated paraffin and 4 in. of lead con-

tained by $\frac{1}{2}$ -in. steel plate. Stepped removable lead sections in the central region of the shield are 5 in. thick and are contained by edge wrappings of $\frac{1}{16}$ -in. steel.

Sectional views of the horizontal thermal column appear in Fig. 1-64. The center line of the smallest removable section of the shield door is 53 in. above floor level. A paraffin-filled box is used as a beam catcher. The square central bar is $3\frac{1}{2}$ in. on each side. The intermediate column, which may be removed, is $10\frac{1}{2}$ in. in cross section. The outer dimensions of the complete assembly are $17\frac{1}{2}$ by $17\frac{1}{2}$ in. The slots which appear at the face of individual pieces provide access to aluminum pulling pins included for handling convenience. Figure 1-65 is an exploded view of all movable components of the column,

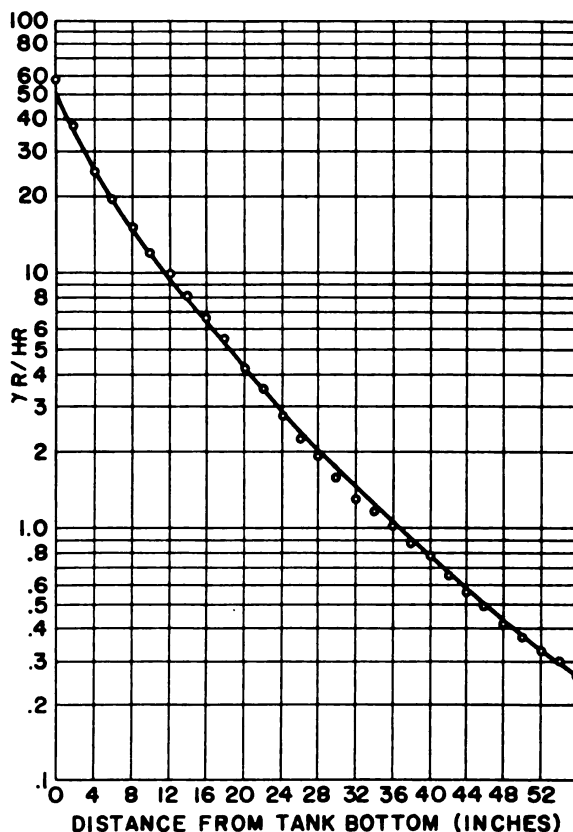


Fig. 1-63 Vertical gamma-ray distribution in water tank.

showing their geometric relationships and sizes. All removable graphite pieces which are located on the center line of the column were duplicated

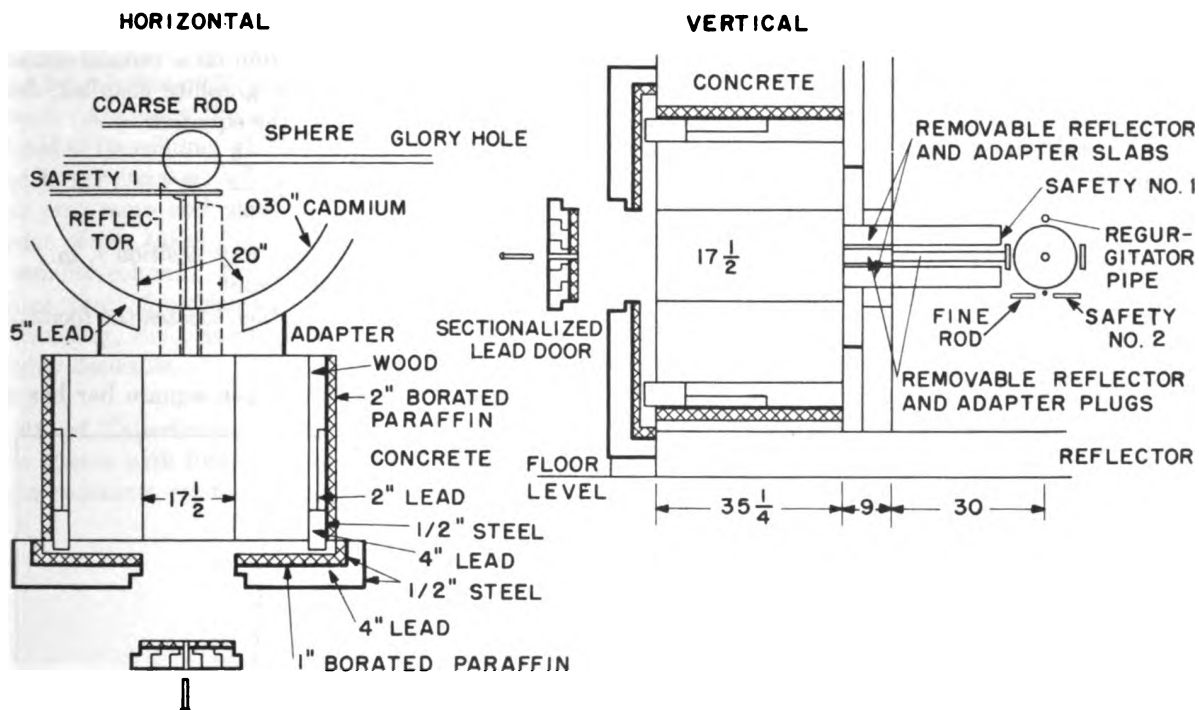


Fig. 1-64 Sections through horizontal thermal column.

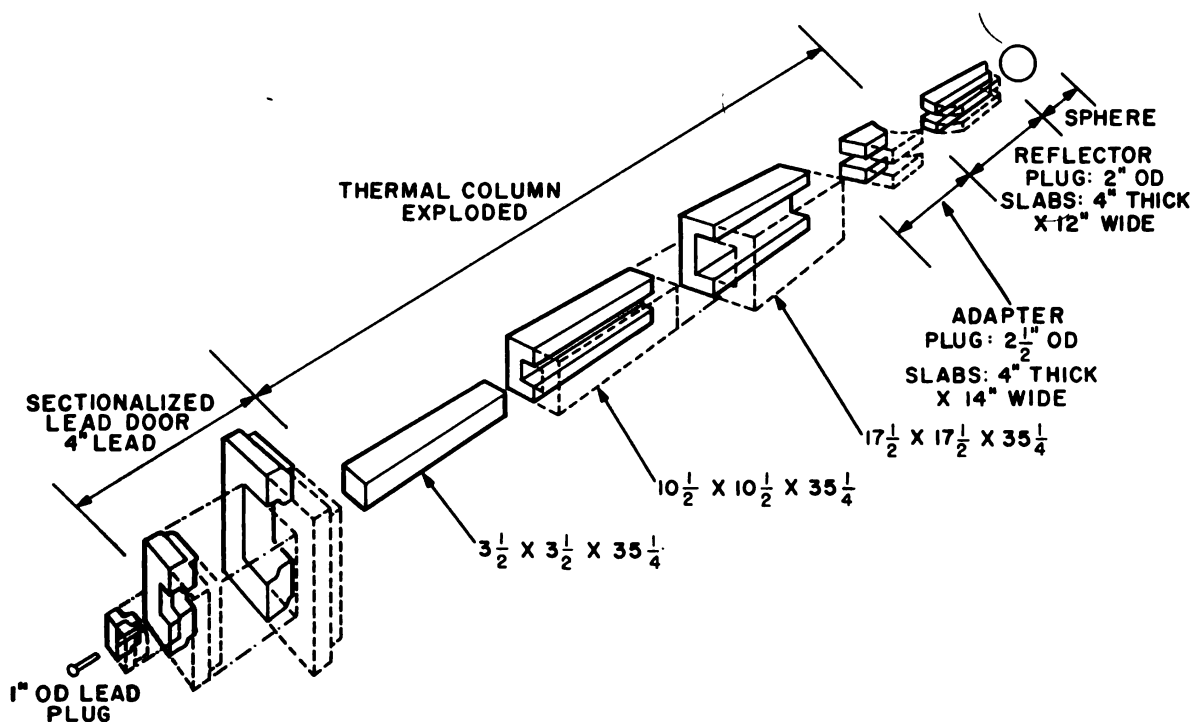


Fig. 1-65 Geometry of removable components of thermal column.

and specially machined for purposes of foil measurements.

The horizontal column is not equipped with a cadmium shutter. The inner face of the gamma-ray shield is lined with borated paraffin, and a cadmium cover is used on the inner faces of the

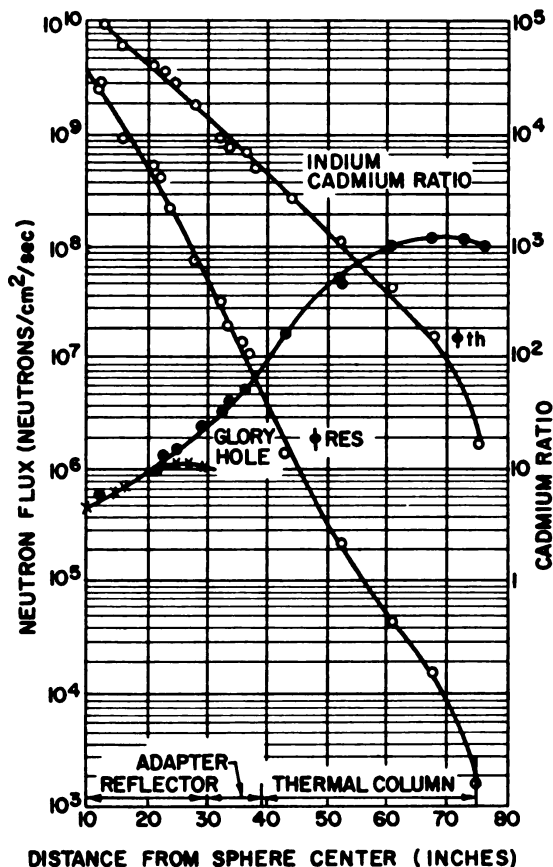


Fig. 1-66 Neutron distribution in horizontal thermal column.

removable lead sections when neutron beams are not desired.

The distribution of neutrons along the center line of the column is indicated in Fig. 1-66. Indium foils, bare and cadmium-covered, were used for the measurements. The cadmium ratio curve demonstrates the finite geometry effects of the thermal column quite clearly. The reduction of the cadmium ratio at the face of the column is seen to be considerably less severe than at the cadmium-wrapped reflector shown in Fig. 1-59.

Figure 1-67 indicates the neutron distribution across the face of the column on a vertical center line. With all removable graphite installed, the function is described by the equation

$$\phi = \phi_0 \frac{r^2}{4a}$$

where ϕ = neutron flux at any position r , in.

ϕ_0 = flux at center point

$a = 7.7 \times 10^{-5}$, which locates the focus of the parabola

Removal of the central $3\frac{1}{2}$ -in.-square bar has a

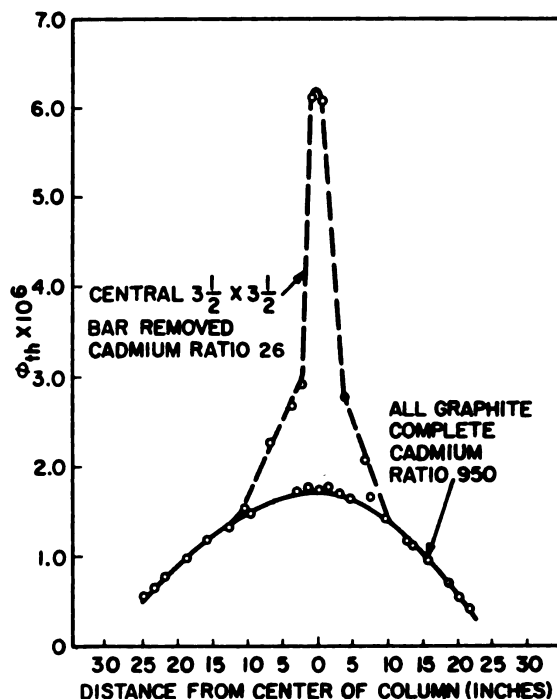


Fig. 1-67 Neutron distribution across face of horizontal thermal column.

marked effect on the neutron outflux in that vicinity, as shown by Fig. 1-67. The cadmium ratio is simultaneously reduced a significant amount.

The diffusion length of thermal neutrons down the center of the column is 21.2 cm. This relatively short diffusion length results from the finite size of the column and consequent edge effects. A column approximately 5 ft square has been found to have a diffusion length of 29.4 cm. From these data it is seen that two columns of equal length having 4- and 5-ft-square cross sec-

tions will differ in usable outflux by a factor of approximately 7 in the case of 5-ft-long columns and by a factor of 25 in the case of 8-ft-long columns. Assuming the cost of a column is proportional to its volume, which is probably reasonable, the larger-cross-section column will cost only 50 per cent more and yield usable outfluxes on the order of ten times the magnitude obtained from a smaller column. From this conclusion it is obvious that, financial and design considerations permitting, the column of larger cross section is highly desirable.

Critical Experiment. This consists of loading the sphere with fuel solution and evaluating certain constants related to the critical mass. The

placed in the geometrical center of the sphere. Uranyl sulfate was added in successively decreasing aliquots following an initial charge of 12 liters of distilled water and a small amount of sulfuric acid. After each aliquot was added, source-multiplication measurements were taken at various positions around the reactor. The neutron detectors used were the three BF_3 tubes on the sides of the reactor, three special BF_3 tubes temporarily laid on the top of the reactor, and seven indium foils spaced along the glory hole. Detector locations are indicated in Fig. 1-68.

Multiplication measurements used to predict criticality were made with all rods completely removed from the reactor. In addition, counting rates were taken with various combinations of rod configurations, as shown in Fig. 1-69. The latter measurements were taken after about 500 g of U^{235} were added, permitting evaluation of the effectiveness of the control and safety rods. Very

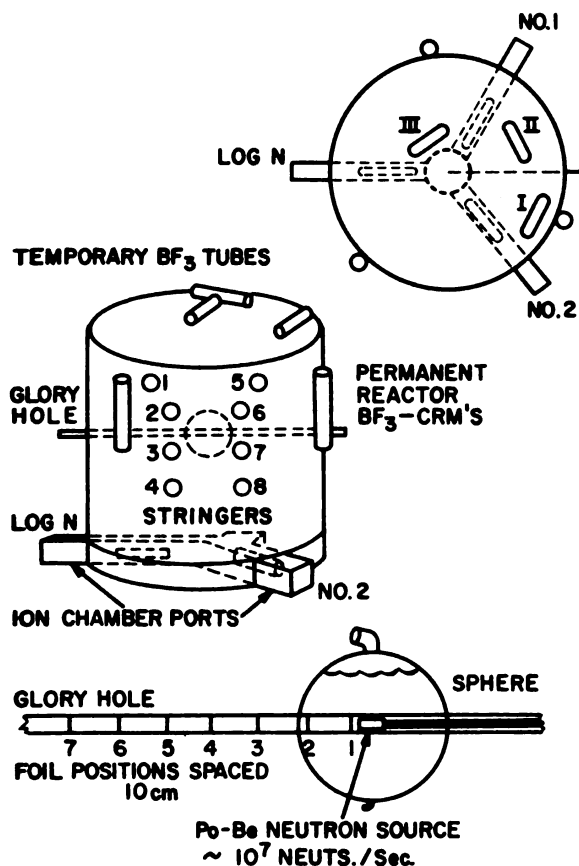


Fig. 1-68 Source and detector locations during critical experiment.

conventional source-multiplication technique was used to determine the critical mass of the boiler. Two 1.5-curie polonium-beryllium sources were

SCHEDULE OF ROD & SOURCE EVALUATION				
NO.	SAFETY RODS	COARSE ROD	FINE ROD	SOURCE
1	IN	IN	IN	CENTER
2	OUT	IN	IN	CENTER
3	OUT	OUT	IN	CENTER
4	OUT	OUT	OUT	CENTER
5	OUT	OUT	OUT	30 CENTIMETERS
6	IN	OUT	OUT	30 CENTIMETERS
7	OUT	IN	OUT	30 CENTIMETERS

*30 CM FROM CENTER OF SPHERE OR ABOUT 6 INCHES INTO REFLECTOR FROM EDGE OF SPHERE.

Fig. 1-69 Schedule of rod and source evaluation.

late measurements taken with the source in the various positions allowed prediction of its poisoning.

Special burettes were designed and fabricated for measuring the volume of fuel added to the reactor. Experience has shown that additions can be made with an accuracy of ± 0.1 ml, which is equivalent to ± 0.02 g of U^{235} . The glass burette was enclosed in a gasketed watertight plastic container to ensure containment of the fuel if the burette should leak or break.

Loading schedule and results. The following carefully planned criterion was to apply during the loading schedule: No more than one-half the amount of U^{235} required to produce a critical con-

dition, according to estimates based on the multiplication measurements, was to be added until this required amount decreased to less than 20 g; after this time no addition was to exceed 10 g of U^{235} , and successive additions were to decrease in

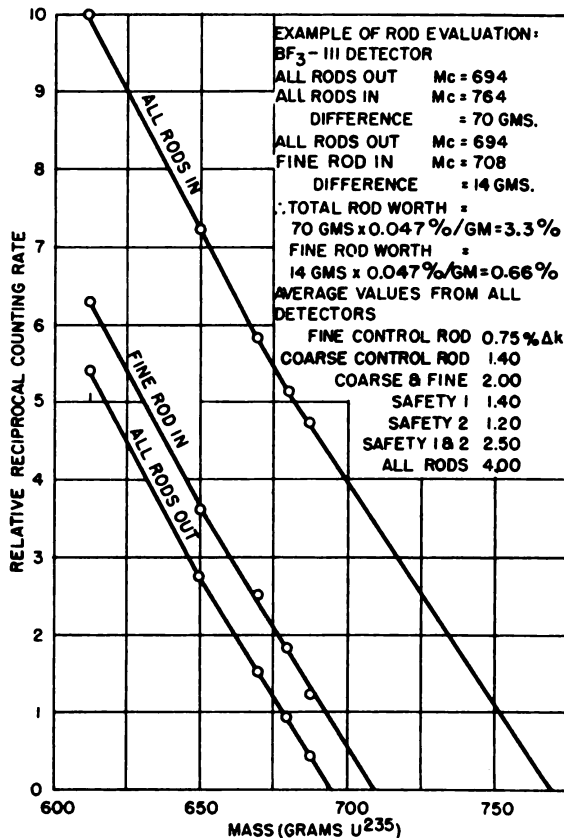


Fig. 1-70 Rod evaluation during critical experiment.

size progressively. This ensured a careful approach to criticality by the addition of less than 0.5 per cent reactivity (the estimated effectiveness of the weakest control rod) in any one aliquot as criticality was neared. If it was found impossible to reach a critical condition with the proposed loading, the concentration would be increased by vacuum-distilling the fuel solution in the mixing bowl.

Detectors were positioned around and in the reactor as shown in Fig. 1-68. Foils were inserted in the glory hole from the west, the neutron source from the east.

The planned loading procedure was followed with few exceptions. The principal exceptions

consisted of the insertion of a temporary cadmium safety rod in the glory hole for aliquot 2, and a reduction in amount of the last few aliquots because data extrapolation indicated a critical mass 6 per cent less than originally anticipated. The temporary safety-rod precaution was taken because extrapolation of aliquot 1 data did not allow the addition of aliquot 2 as planned if the rule of adding less than one-half the amount necessary for criticality was to be observed. This exception was felt justified since available criticality data showed the extrapolation to be overly conservative.

Commencing with aliquot 5, all the permutations of rod and source configurations shown in Fig. 1-70 were used to obtain multiplication data.

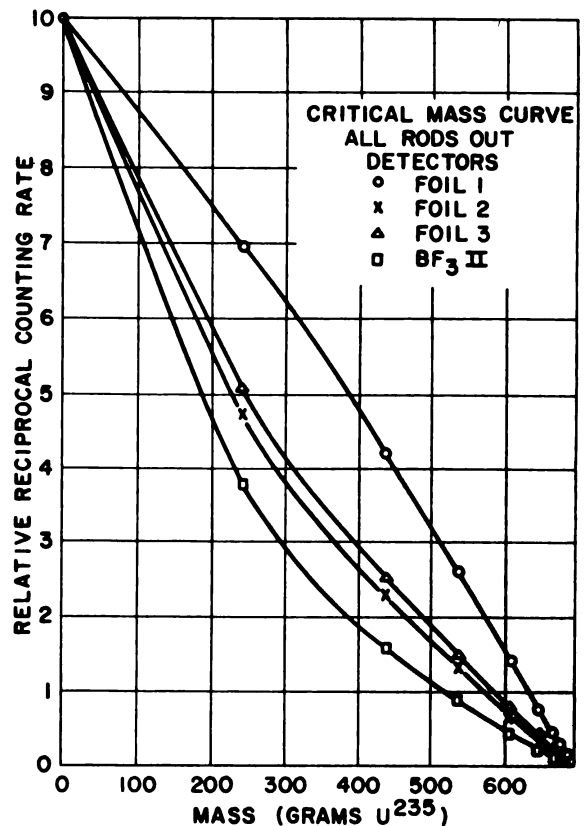


Fig. 1-71 Critical-mass curve, all additions.

Examples of these are plotted in Fig. 1-69. These data allowed continuous evaluation of rod effectiveness, in terms of grams of U^{235} in the sphere as well as in the final evaluations in per cent k .

Figures 1-71 and 1-72 show the reciprocal multiplication curves given by various detectors with all rods removed. Figure 1-72 was used during later additions, and it may be seen that extrapolations became consistent after aliquot 5 to a value of approximately 694 g.

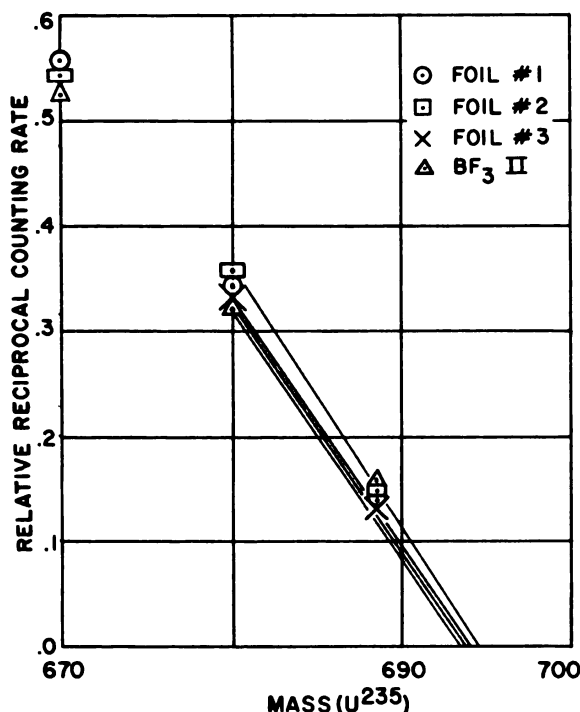


Fig. 1-72 Critical-mass curve, last additions.

Figure 1-73 shows a plot of counting rate on a BF₃ monitor as a function of time during a loading and mixing cycle for aliquot 2. This demonstrates the necessity for mixing the solution.

After the addition of aliquot 8 and withdrawal of the source container, it was found that the reactor was capable of rising on a 68-sec period with all rods out. This period corresponds to an excess reactivity of 0.111 per cent. It may be shown by the inhour equation that the critical mass of the reactor at this temperature and loading was 685.4 g. Comparing the latter value with the extrapolation data, it becomes apparent that the source capsule was equivalent to approximately 8 g of U²³⁵.

Radiation levels outside the biological shield were negligible during the precritical phases of the loading. During period measurements, toler-

ance levels were never exceeded outside the biological shield.

After criticality had been achieved, there remained several small additions to be made to the fuel solution. It was felt that the pH of the solution should be lowered to less than 1.0 to prevent precipitation of uranium over long periods of time.

Accordingly, a total of 150 ml of concentrated sulfuric acid was added in three aliquots and resulted in a final pH of 0.7. The first such addi-

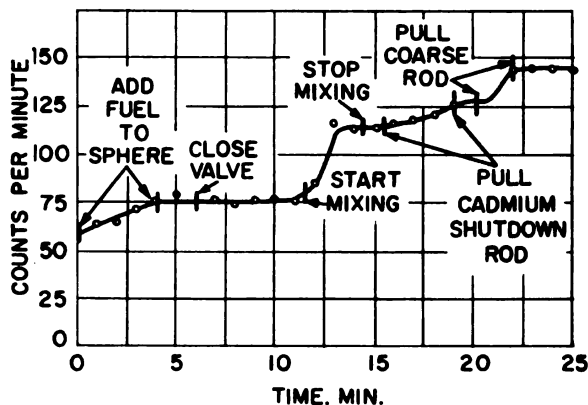


Fig. 1-73 Counting-rate change during aliquot 2.

tion resulted in an increase in reactivity, and the following two aliquots had a negative reactivity effect. This effect may be qualitatively interpreted on the basis that the reactivity is increased by more completely filling the sphere, improving the geometry and thus decreasing leakage, and at the same time the reactivity is decreased by dilution and poisoning. These additions and their effects are quantitatively described below, under Critical Mass Calculations. The discrepancy between excess reactivity before acid additions given there (0.074 per cent) and the value shown above (0.111 per cent) is explainable by a slight temperature change of the solution. All calculations given there are based on the 0.074 per cent value because it was determined just prior to the postcritical additions and at what is considered the standard temperature.

Excess reactivity for experimental purposes was obtained after the pH adjustment by the addition of 6.4 g of U²³⁵ in two aliquots; the effects will be described later. The solution level after all addi-

tions results in an unoccupied volume of 300 ml at the top of the sphere. This volume permits bubbling and thermal expansion without fuel loss from the sphere.

The operating mass of the reactor is 694.2 g of U^{235} . This mass gives a minimum period at room temperature of 13 sec, which corresponds to an excess k of 0.3 per cent.

Critical mass calculations. After aliquot 8, the reactor contained 687.8 g of U^{235} , and a period of 68 sec was measured with the source removed. This corresponds to an excess reactivity of 0.111 per cent. Excess reactivity was computed from the following equation.

$$\Delta k = \left(\frac{135}{T} + \frac{626}{T + 0.62} + \frac{6287}{T + 2.19} + \frac{16,538}{T + 6.51} + \frac{62,604}{T + 31.7} + \frac{24,421}{T + 80.2} \right) \times 10^{-4}\%$$

where T = period, sec

Δk = excess reactivity, per cent

$\Delta k = (k_{eff} - 1)100$

Using the previously calculated mass coefficient of 0.046 per cent/g, this corresponded to an excess mass of 2.4 g, giving the critical mass as 685.4 g. Extrapolations of the subcritical multiplication data show a critical mass of 693.4 g of U^{235} with the source in the center of the sphere. Therefore, the source was worth 8.0 g (693.4 - 685.4), or 0.37 per cent Δk .

After aliquot 8, the source was moved to a recess in one of the lower stringers and permanently mounted there.

After several days of preparation, the fuel loading was resumed. At this time, a period of 116 sec was measured, corresponding to a reactivity of 0.074 per cent, or 1.6 g excess. Therefore, the critical mass at this point was 687.8 - 1.6, or 686.2 g. The difference of 0.8 g from the previous value is accountable by temperature change.

Counting rates were taken with the fine rod in, to keep the reactor subcritical. This procedure was followed during the remainder of the additions.

The period of 116 sec gave an excess reactivity, Δk , of 0.074 per cent, or a k_{eff} of 1.00074. Extrapolations of the subcritical data gave an estimate of the fine rod's reactivity value as 0.75 per cent.

Then k_{eff} with the fine rod in is 1.00074 - 0.0075, or 0.9932. This corresponds to a multiplication of

$$M = \frac{1}{1 - k_{eff}} = 148$$

The effect of succeeding aliquots was determined by the net increase or decrease in multiplication. To do this, a derivation is presented, giving the new k_{eff} resulting from an addition where the multiplications and k_{eff} values before and after the addition are M_0 and M_1 , and k_0 and k_1 , respectively.

$$M_0 = \frac{1}{1 - k_0} \quad M_1 = \frac{1}{1 - k_1} \quad \frac{M_0}{M_1} = \frac{1 - k_1}{1 - k_0}$$

$$1 - k_1 = \frac{M_0}{M_1} (1 - k_0) \quad k_1 = 1 - \frac{M_0}{M_1} (1 - k_0)$$

Sulfuric acid additions. The first aliquot of sulfuric acid was 40 ml. The resulting multiplication was 152. By computation from the above formula, $k_{eff} = 0.9934$ with the fine rod in. Adding in the fine-rod value of 0.0075 gives $k_{eff} = 1.0009$. At this time the fine rod was withdrawn and a period of 112 sec was measured, corresponding to a Δk of 0.076 per cent, giving $k_{eff} = 1.0008$.

A 70-ml aliquot of sulfuric acid was then added with a resulting multiplication of 139 corresponding to a k_{eff} of 0.9928 with the fine rod in. Adding in the fine-rod worth gives $k_{eff} = 1.0003$. A period of 421 sec was measured. This corresponded to an excess reactivity of 0.024 per cent, or a k_{eff} of 1.0002.

The final aliquot of sulfuric acid was 40 ml, and this gave a multiplication of 132 corresponding to a k_{eff} of 0.99993 with the fine rod out, and the reactor was thus slightly subcritical.

Therefore, the net effect of adding the sulfuric acid was 0.9993 - 1.00074, or 0.00081, or a reactivity loss of 0.081 per cent, which is approximately equivalent to 1.76 g of U^{235} .

At this point preparations were made to add an excess reactivity of 0.3 per cent for experimental purposes. Again counting rates were taken with the source in stringer 8, and criticality was prevented by keeping the fine rod in.

Multiplication measurements taken before adding the first excess aliquot indicated an increase

in reactivity corresponding to a temperature decrease. Multiplication was 137, and k_{eff} was thus 1.0002. This corresponds to an excess mass of 0.43 g, and so predicted critical mass at this point was 687.3 g.

The first aliquot was 3.4 g of U^{235} . The multiplication rose to 173, and k_{eff} increased to 1.0017. A period measurement of 42 sec gave a k_{eff} of 1.0016. Thus, the net effect of this addition was to add an excess reactivity of 0.15 per cent.

The final addition of 3.0 g of U^{235} increased the multiplication to 238, giving a k_{eff} of 1.0032. At this time a period of 13.3 sec was measured, corresponding to a k_{eff} of 1.0029.

Therefore, the excess reactivity is equal to 0.30 ± 0.02 per cent.

Total loading was 694.19 g. The previously estimated critical mass was 687.34 g at the start of the excess loading. Therefore, the excess mass added is $694.19 - 687.34$, or 6.85 g of U^{235} . Then

$$\frac{\Delta k}{\Delta M} = \frac{0.30}{6.85} = 0.044 \pm 0.006\%/g$$

The error in M is estimated to be ± 0.46 g. Therefore, the critical mass equals 687.3 ± 0.5 g of U^{235} .

Operating Characteristics. Neutron intensities in horizontal thermal column. A series of experiments was conducted to determine the optimum graphite configurations in the horizontal thermal column for external neutron-beam production. Seven cases were investigated and are described in Table 1-11.

Table 1-11

Case number	Graphite condition
1	All graphite in place
2	3½-in. bar removed
3	3½-in. bar, adapter and reflector plugs removed
4	10½-in. bars removed
5	17½-in. bars removed
6	17½-in. bars, adapter slabs, and plugs removed
7	All removable graphite withdrawn

Maximum thermal-neutron beam intensities at the outer surface of the lead shield and 32 in. from the shield surface were measured with standardized bare and cadmium-covered indium foils. The lead shield was complete in all cases tested.

The shield itself reduces the neutron flux appearing at its outer surface by a factor of about 10 with all graphite in place (cadmium ratio = 10^3).

Intensity of the thermal-neutron beam at the two experimental locations is shown in Fig. 1-74.

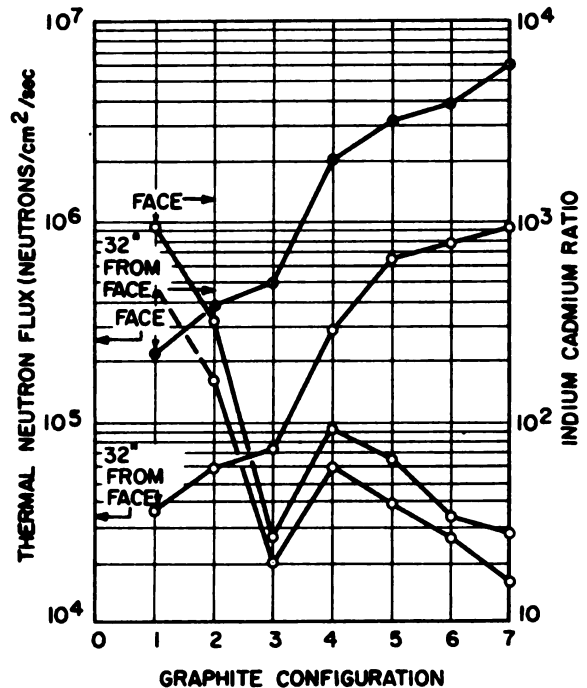


Fig. 1-74 Horizontal-column thermal-neutron beam intensities and cadmium ratios.

It is clearly indicated that the cavity formed by removing graphite from the column represents considerable assistance to beam production, and the increase in beam intensity is very nearly proportional to the increase in cavity cross-sectional area. Presented on the same graph are indium-cadmium ratios observed with the various graphite loadings. In comparing both sets of curves it has been concluded that configuration 4 probably represents the optimum for general thermal-neutron beam experiments. The conclusion is based on availability of a reasonably pure and intense beam of thermal neutrons with a minimum of attendant fast neutrons and gamma rays.

Auxiliary bismuth shielding blocks have been employed on occasion. It has been determined that 6 in. of bismuth placed adjacent to the adapter graphite and having the full cross section of the cavity in configuration 4 reduces the thermal-

neutron beam by a factor of less than 2 in comparison to no bismuth.

D₂O-filled aluminum tanks were inserted in the cavities formed by configurations 4 and 5. The D₂O was not of high purity as it contained 2.2 per cent H₂O. Reduction of thermal outflux by a 12-in. thickness of D₂O of full-cavity cross sec-

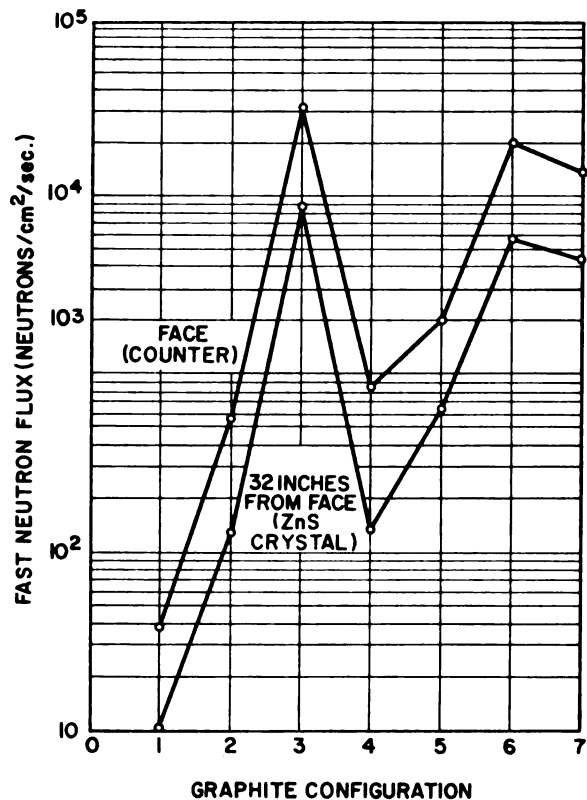


Fig. 1-75 Horizontal-column fast-neutron beam intensities.

tion ranged from factors of approximately 5 at the face to 2, depending on the depth of insertion into the column.

Fast-neutron production has been measured in the same locations and for the same graphite configurations as were considered for thermal-neutron beams. The instruments used were a silver-activated zinc sulfide crystal-photomultiplier arrangement reporting to a scaler, and an air-equivalent proportional counter. The crystal detector was located 32 in. from the shield-face position, and the counter was used for measurements directly against the shield. As in the thermal-neutron tests, the shield door was complete. The data

are presented in Fig. 1-75. The values for configuration 7 should probably be the highest reported. This discrepancy is unresolved.

Gamma-ray contamination of the neutron beams has been measured for some of the graphite configuration cases, but inconclusive results have been obtained. The maximum gamma-ray intensity at the face of the graphite has been determined as 9.3 r/hr at the 500-watt power level. The spectrum of those gamma rays above 1 Mev has been measured and reported; no detectable lines are present. The lead shield with cadmium lining provides a reduction factor of about 260 in the gamma-ray intensity. The intensity at the face of the adapter graphite is approximately 450 r/hr with configuration 5.

The horizontal thermal column, properly equipped with removable inserts, represents an extremely versatile experimental facility. The gamma-ray shield at the face of the column was designed for a 100-watt power level and so fabricated that additional lead could readily be installed to permit higher power levels. The shield has performed as expected and operation at 500 watts does not produce significant personnel radiation hazard. The borated paraffin used appears to function satisfactorily since no measurable reduction of emergent gamma rays at the outer face of the shield was effected by the addition of a layer of cadmium. No apparent loss of neutrons results from the rather severe cross-section reduction at the reflector-column interface. Little or no neutron interaction between the vertical and horizontal columns is expected because of the cadmium wrapping present at the reflector surface.

Temperature coefficient of reactivity. The temperature coefficient has been measured between 10.8 and 37.6°C. All measurements were made using the following procedure:

1. The fuel-solution temperature was adjusted to the desired value. This was done by using the cooling system to heat or cool the core by passing the circulating water through a heater or refrigeration unit.

2. The temperature was allowed to equalize throughout the sphere by a several-minute wait before the reactivity measurement. No detectable temperature change in a period of 5 min was considered sufficient evidence of temperature sta-

bility. Cooling- or heating-water circulation was normally discontinued 10 min before the reactivity measurement. The temperature, as reported by a multipoint recorder, could be read with an accuracy estimated at $\pm 0.1^\circ\text{C}$.

3. Safety and control rods were withdrawn.

4. The power level was allowed to rise a minimum of 1 decade after the rods were completely removed from the reactor before period measurements began.

5. A set of three "standard" periods was taken for each temperature. The three periods were determined at different ranges of power during the same power rise. A logarithmic amplifier and recorder, operated by a gamma-compensated ionization chamber, were used to indicate power level. The standard periods were measured between 1 to 50, 5 to 100, and 10 to 100 watts.

6. The period measurements were converted to Δk values by using the water-boiler inhour formula:

$$\Delta k = \left(\frac{135}{T} + \frac{626}{T + 0.62} + \frac{6287}{T + 2.19} + \frac{16,538}{T + 6.51} + \frac{62,604}{T + 31.7} + \frac{24,421}{T + 80.2} \right) \times 10^{-4}\%$$

If the first period measured (1 to 50 watts) departed significantly from the two later measurements, it was discarded on the basis that equilibrium of the delayed neutrons had not been achieved by the time the measurement began.

7. Periods to be measured were restricted to a minimum of about 13 sec, with two exceptions, necessitated by data normalization. It was felt that restriction to the longer periods was necessary if equilibrium was to be attained because of the relatively long time required to withdraw the control rods. In a range of fuel temperatures between 10.9 and 16.7°C , a boron absorber was introduced in the central exposure facility to decrease the available reactivity. Similarly, a solid rod of Lucite and a D_2O -filled aluminum tube were used successively for temperatures above 27.2°C in order to increase the available reactivity. It was found that this method of adjusting the reactivity was acceptable since the slope of the reactivity-temperature curve was not affected. Figure 1-76 demonstrates this point graphically and also presents the total data gathered.

Power-temperature oscillations. In the second experiment performed, the reactor was allowed to rise in power to a level of 500 watts. No cooling water was circulated through the core. The

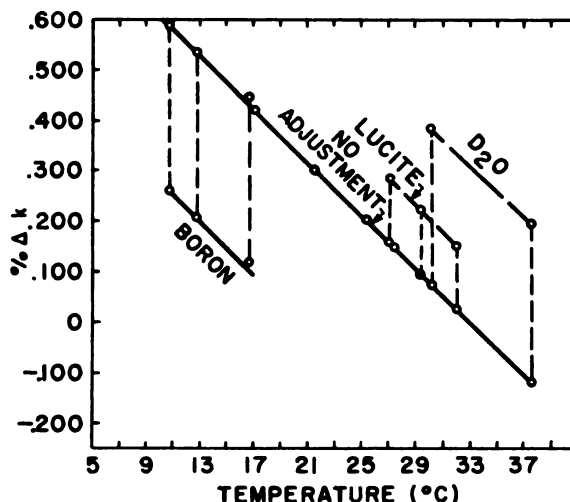


Fig. 1-76 Reactivity vs. temperature.

power level of 500 watts was maintained for approximately 1 hr by the automatic control system, until the reactor was no longer capable of remaining critical with all control rods removed because of the fuel-temperature rise from power dissipation in the core. At this time, with all

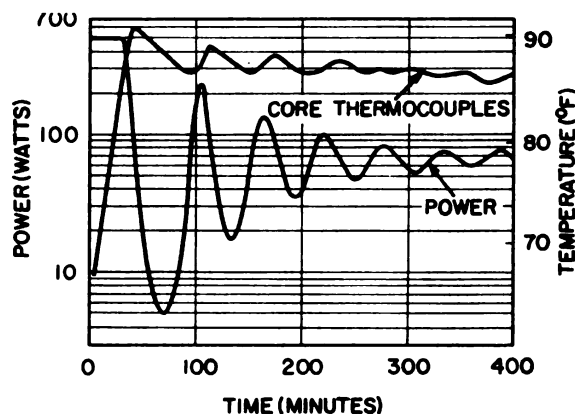


Fig. 1-77 Power, temperature vs. time oscillation experiment.

rods removed, the reactor power was allowed to begin oscillating as the core was alternately cooled by conduction and heated by power dissipation. The power and core temperature were

noted as a function of time. Figure 1-77 is a graphical presentation of the data. Temperature data in this experiment were taken with the thermocouples mentioned previously.

Power transients. Having determined the temperature coefficient, it was of interest to see if it was possible to predict, in the simplest case, the peak power attainable in an unperturbed, except for temperature increase, rise in the boiler power. This point is of particular interest in regard to reactor-hazard calculations. Although the 500-watt reactor, most particularly the biological

Gas flow to the recombiner, caused by unequal pressure distribution in the sweep-gas system, assured partial recombination of the hydrogen and oxygen evolved. Explosion traps and the absence of any ignition source negated the possibility of damage from inflammation or detonation in the localized regions where hydrogen concentration could conceivably reach the required values. In addition, the sphere assembly and gas system were designed and fabricated to withstand a pressure of 300 psi, which is in excess of the maximum pressure developed in any hydrogen-oxygen deto-

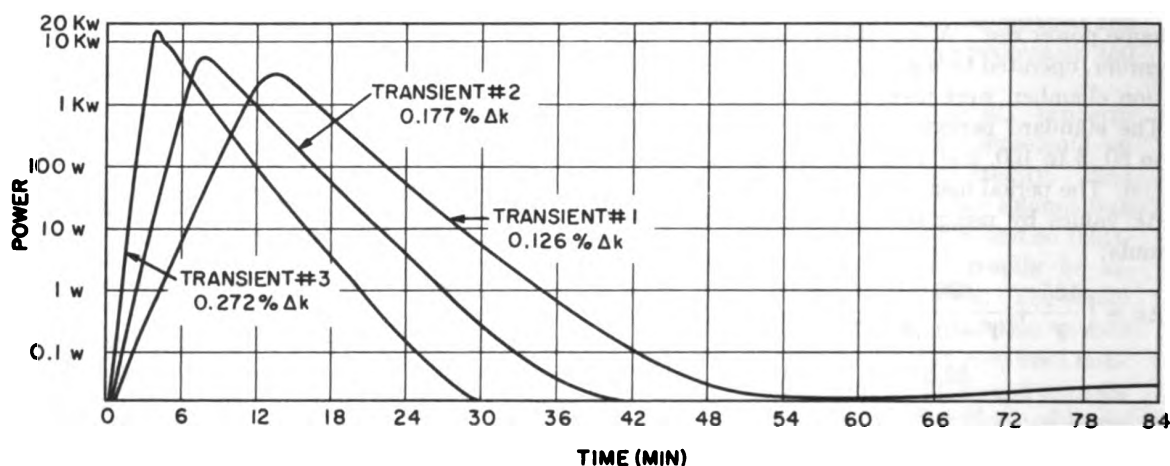


Fig. 1-78 Power vs. time-transient experiment.

shielding, is not designed for prolonged operation at power levels in excess of 2 kw, it was determined that a series of low-excess-reactivity excursions was entirely practicable. Consideration was given to the possibilities of physical abuse to the reactor and its auxiliaries and to personnel exposure to radiation. The spillover tank, which is attached to the core and acts as a reservoir for expelled fuel solution, assured no ultimate change in the core loading. With the reactor sweep-gas static pressure adjusted to a sufficiently low value before each transient, the large volume of the gas system and the presence of the catalytic recombiner in the gas system avoided the possibility of positive pressures greater than a few inches of water gauge being developed in the closed-cycle gas system. The experiments were performed without operation of the sweep-gas system since it had been previously observed that this affected the rate of heat loss from the sphere.

nation. Personnel exposure was predicted and found to be less in all cases than that received by the operating personnel during 1 hr of normal operation at 500 watts.

After the analysis of potential hazards, it was concluded that a series of three transients, in order of increasing reactivity, should be studied. Estimates had been made of the peak power to be expected from transients resulting from excess reactivities of 0.1, 0.2, and 0.3 per cent. The results were found to be 1.96, 10.3, and 31.1 kw, respectively. The calculations were based on the assumptions that the sphere was isolated from the graphite reflector; that the heat capacity of the core includes the fuel solution, the part of the cooling coil immersed in the fuel solution, the glory-hole tube, and approximately 6 in. of the 2-in.-diameter pipe fastened to the top of the sphere. The heat capacity of these components is 16.6 kcal/°C.

The transients measured and recorded involved excess k 's of 0.126, 0.177, and 0.269 per cent. The first case involved a slightly greater excess reactivity than desired because of imperfect temperature adjustment of the fuel solution. The latter two cases exhibited the presence of the phenomenon described above. Beforehand knowledge of the exact excess k was not important as long as it was reasonably close to the desired value, since it could be determined from the data taken. Peak powers attained were 3.1, 5.8, and 15.8 kw, in order of greater excess k 's. Peak-power predictions of transients involving these exact excess k 's, made with the assumptions mentioned previously, yield values of 3.19, 7, 8, and 20.8 kw. Figure 1-78 is a plot of the three transients based on a common time scale.

From mechanical integration of the power transient curves the information in Table 1-12 has been established.

Table 1-12

	Transient		
	No. 1	No. 2	No. 3
Total energy, watt-hr.....	261	335	517
Energy to turnaround, watt-hr.....	109	108	128
Energy to departure from initial period, watt-hr.....	6.8	13.6	27.7
Miscellaneous information:			
Δk , %.....	0.126	0.177	0.269
Period, sec.....	57	34	16
Peak power attained, kw...	3.1	5.8	15.8
Temperature rise of core, °C.....	5.3	8.1	13.0

Static pressure in the gas system was measured before and after each transient. It was thought that a measure of gas production as a function of pressure under transient conditions might be obtained, depending upon the action of the recombiner. However, the catalytic recombiner interfered with the measurement by recombining significant portions of the hydrogen and oxygen.

Power calibration by foil techniques. A number of $\frac{1}{2}$ -in.-diameter foils were punched out of 0.0025-in.-thick gold sheet and were cemented on

1-in.-OD by $\frac{1}{16}$ -in.-thick Lucite disks. The foils were then irradiated in a calibration tank at equal distances from a neutron source and counted. By comparing the saturated activities, correction factors were assigned to each foil to compensate for slight variations in foil thickness, density, impurities, and mounting imperfections.

Foils were placed inside a 2-in.-thick lead pig facing a thin end-window Geiger tube. This feeds a decade scaler which counts the foils for a predetermined time or number of counts. The resulting activities are corrected, by conventional means, for background, foil factor, coincidence loss, and daily variations.

Four of the gold foils were irradiated in a known flux and cadmium ratio. It was possible to relate the thermal activity of a gold foil (bare-foil activity minus cadmium-covered activity) to the thermal flux incident upon that foil. It has been determined that 3.60 ± 0.07 neutrons/cm²/sec are equivalent to 1 count/min corrected saturated activity.

To determine the thermal-neutron flux in a given position, two foils must be irradiated. One of these is a bare foil mounted as previously described. The other is an unmounted foil enclosed in a 0.030-in.-thick cadmium box. The thermal activation is consequently the difference between the two foil activities.

A series of indium and gold foils were exposed within the sphere at varying distances from the center. Their activities normalized to the calibrated 500 watts are shown in Fig. 1-79. Also shown are the barium fission-product activities from uranium foils exposed in various locations.

An equation of the following form has been fitted to the curve as a solution to the neutron distribution in the core:

$$\phi = \phi_0 \frac{\sin kr}{kr}$$

where $k = 0.31$ in.

r = distance from center of sphere, in.

The accuracy of this curve fitting is shown in Fig. 1-79.

Integration of this equation describing the neutron distribution over the volume of the sphere gives a ratio of average to peak neutron, or power density in the core, of 0.819.

The power is defined in the following manner:

$$P = \bar{\phi}_{th} \Sigma_{f25} CV$$

where P = power, watts

$\bar{\phi}_{th}$ = average thermal-neutron flux in core

Σ_{f25} = macroscopic thermal-fission cross section for U^{235}

C = conversion factor to convert fissions per second to watts

V = volume of fuel solution, cm^3

Since Σ_{25} , C , and V are known and the thermal flux may be determined by exposing bare and

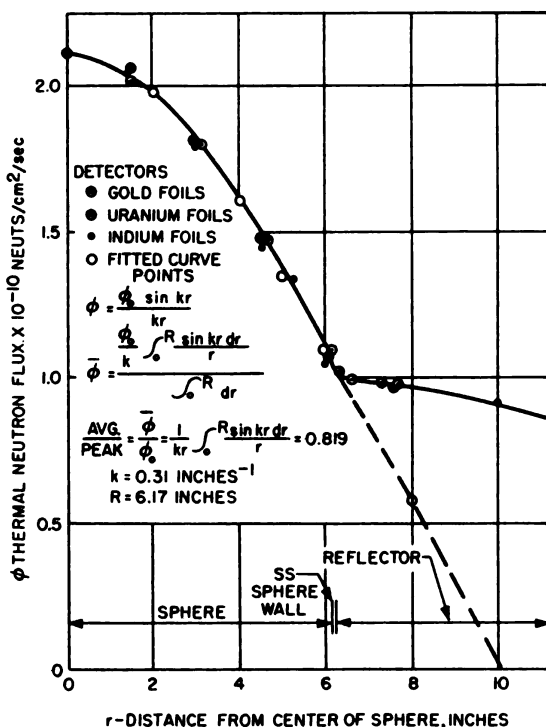


Fig. 1-79 Flux traverse in core.

cadmium-covered gold foils, the power can be determined readily from the above expression. A solution of this equation shows that a flux of 4.22×10^7 neutrons/ cm^2 /sec ± 10 per cent at the center of the sphere is equivalent to 1 watt. In this solution the fission cross section of U^{235} is assumed to be 549 barns, the volume of fissionable solution is about 14 liters, and the conversion factor is 3.7×10^{10} fissions per second per watt, assuming the energy release is 175 Mev per fission.

A series of 12 gold foils, alternately bare and cadmium-covered, were exposed in the center of the sphere at approximately 1 watt, and they were counted using the previously described techniques. Corrected saturated activities were computed and converted to thermal activities and thermal fluxes. The resulting power was 1.30 ± 0.14 watts. Several days later a series of six bare gold foils was exposed in the center of the sphere, and again the power was calculated using the cadmium ratio of 3.24 ± 0.09 as measured by gold, which was determined by the preceding run. The result was 1.31 ± 0.14 watts. It was felt that this method had proved feasible subject to the degree of its agreement with the other methods of power measurements.

All irradiations of these foils were made at a power of approximately 500 watts. A series of enriched U^{235} foils were prepared by sealing a measured amount of uranium oxide powder inside a 0.001-in.-thick aluminum foil. They were taped to Lucite backings 1 in. in diameter by $\frac{1}{16}$ in. thick. The quantity of uranium in each foil was about 1 mg.

The foils were exposed to approximately 1.5×10^{13} nvt. The foils were then disassembled, and using standard carrier techniques, the zirconium 97 atoms formed as fission products were extracted and counted relative to a standardized 4π counter. As a further refinement of the technique, barium 139 was extracted and counted, in addition to zirconium 97 for traverse data. The accuracy of this method is estimated to be within ± 10 to 15 per cent.

Three foils were exposed successively in the center of the sphere at approximately 500 watts, and they gave an average flux of 2.45×10^{10} neutrons/ cm^2 /sec, which gives a power estimate of 434 watts ± 12 per cent.

In addition a number of foils were irradiated within the sphere at varying distances from the center. The results are plotted in Fig. 1-79 with the gold and indium traverse points.

Power calibration by calorimetry. The most straightforward method for measuring the power of this reactor is to make a calorimetric determination. However, some difficulties are encountered at this low power since small temperature differences exist in the coolant inlet and outlet lines for reasonable flow rates.

The usual equation defining specific heat governs this application:

$$P = KWc_p \Delta t$$

where P = power, watts

W = mass flow rate, lb/min = (density, lb/ft³) \times [(volume, ft³)/min]

c_p = specific heat, $\frac{\text{Btu}}{\text{lb}^\circ\text{F}}$

Δt = temperature difference between inlet and outlet coolant, $^\circ\text{F}$

K = conversion constant, watts/Btu/min

Copper wells for the temperature-measuring devices were inserted into the flow path of the coolant at the inlet and outlet lines outside the reflector, about 3 ft from the core.

A flowrator (capacity 0.15 to 1.81 gpm, least count 0.002 gpm) was calibrated volumetrically prior to its installation, and a deviation curve was prepared.

Flow rates were usually established at about 0.4 gpm giving temperature differences of about 8°F at 500 watts.

Before taking data, the temperature of the refrigerated heat exchanger was adjusted so that the temperature of the fuel coincided with the ambient temperature to minimize heat loss to the graphite reflector.

Two types of temperature-sensing devices were used: two sets of iron-constantan thermocouple wires were connected in opposition to read temperature difference directly, and a potentiometer was used to measure the emf generated.

The other temperature-measuring device consisted of two thermistors. They were calibrated against precision thermometers, and their resistances were measured on a Wheatstone bridge.

The average power as determined by thermocouple measurements was 503 ± 50 watts. Thermistor readings gave an average power of 509 ± 25 watts for detector 2 setting of 100 on scale 200.

In the case of the calorimetric measurements, the thermocouples and thermistors were carefully calibrated over the full range used in order to minimize this possible source of error. The flowrator also was calibrated over its full scale. Loss of heat to the graphite reflector was minimized by maintaining the core at approximately ambient temperature. The largest source of error in this

method is the change in properties of the coolant water due to its circulation through the cooling circuit. Distilled water was used, but visual observation indicated that impurities were present after many hours of circulation.

The final power results are tabulated below and are shown graphically in Fig. 1-80. Taking all

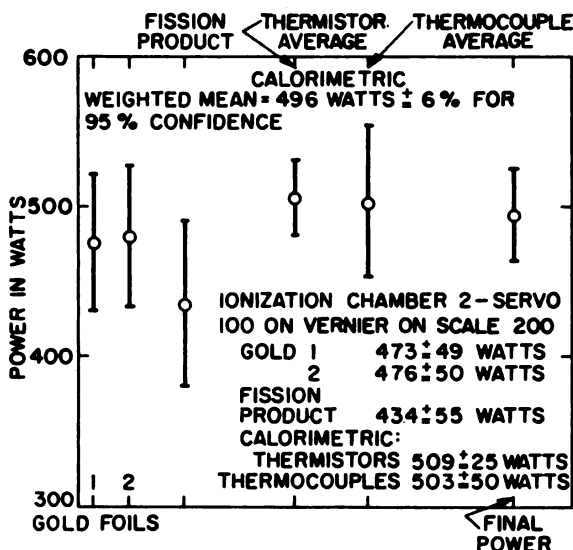


Fig. 1-80 Power levels at standard detector setting.

methods into consideration, the power for a setting of 100 on the vernier, scale 200, on detector 2 (servo) and 25.2, scale 500, on detector 1 has been determined to be $494 \text{ watts} \pm 6 \text{ per cent}$ for 95 per cent confidence limits. The method involving fission-product analysis gives the maximum deviation from the above power, or about 10 per cent lower.

More weight was attached to the thermistor measurement than any other, since it was felt that this was the most inherently accurate method of all because it involved the fewest variables.

Volatile activity build-up and decay. The gross gamma activity found in a sweep-gas line as a function of time from startup and shutdown has been measured. It is possible, with proper valve adjustment, to pass a side stream of the circulating gas through a line external to the biological shield. A length of clean tubing was installed to serve as this external line, the gas pump turned on, and a commercial gamma-ray detector positioned near the line. The reactor was started up and equilibrium power was reached two minutes

after activity was detected in the test line. The output of the detector was fed to a d-c amplifier, which in turn drove a strip-chart recorder. The system was calibrated with a Co^{60} source before the experiment was done.

The detector was an ionization chamber which has a logarithmic response and an essentially flat energy response from 80 kv. A lead half-thickness measurement made on gas-system components approximately 2 hr after shutdown indicated an average energy of about 700 kv. Tygon

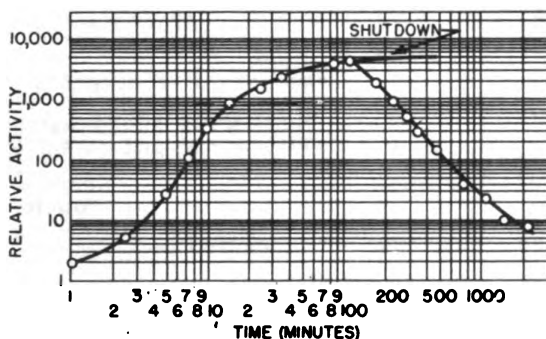


Fig. 1-81 Volatile activity build-up and decay.

tubing $\frac{1}{4}$ in. by $\frac{1}{16}$ in. wall thickness was used as the external line.

Figure 1-81 shows the build-up of the activity in this line and also the decay of the gross products, measured after the reactor was shut down and the gas pump turned off.

Activity of gas-system components. The longer-lived gamma-activity distribution in various components in the closed-cycle sweep-gas system has been measured. The information is presented in Fig. 1-82, which is a schematic diagram of the system. All components are of stainless steel and have wall thicknesses of $\frac{1}{16}$ to $\frac{1}{8}$ in. The activities are given as maximum surface readings in each case. It is generally true that the activity present at any component is proportional to the volume of that component, regardless of location in the system.

Neutron distribution in core and reflector. The flux of slow (<0.3 ev) and indium-resonance neutrons as a function of position in the reflector and core are presented graphically in Fig. 1-83. The

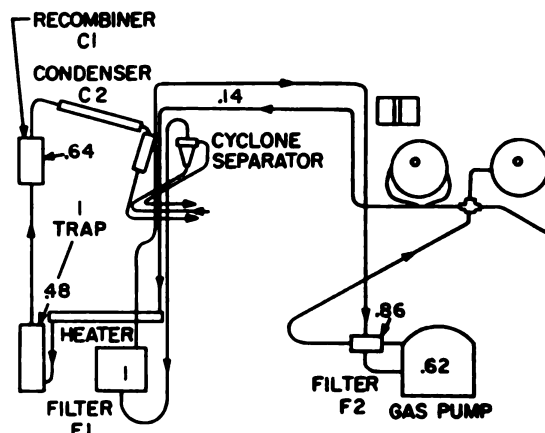


Fig. 1-82 Relative gamma-activity distribution in sweep-gas system.

measurements were made with indium and gold foils, bare and cadmium covered. The cadmium ratio varies from 3.0 at the center to 3.3 at the edge of the sphere and is 13.0 near the edge of the reflector. The actual flux values given are accurate to ± 10 per cent for a 500-watt power level.

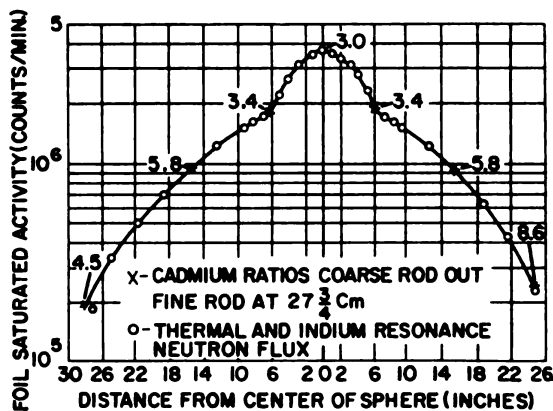


Fig. 1-83 Neutron distribution in core and reflector.

Gamma-ray level in center of sphere at 500 watts. The gamma-ray detector used in the gas-build-up and decay measurements was modified to fit the glory hole (1.08 in. ID) and to minimize neutron response. Its range was extended to 10^5 r/hr.

The body of the modified chamber consisted of a cylinder of graphite capped at each end with a disk of graphite. The electrometer tube was press-fit through one of these disks. Slow-neutron response to a beam of approximately 10^7 neutrons/cm²/sec in a field of 25 r/hr was less than 5 per cent of the total. It was found that prolonged exposure in the sphere at higher powers

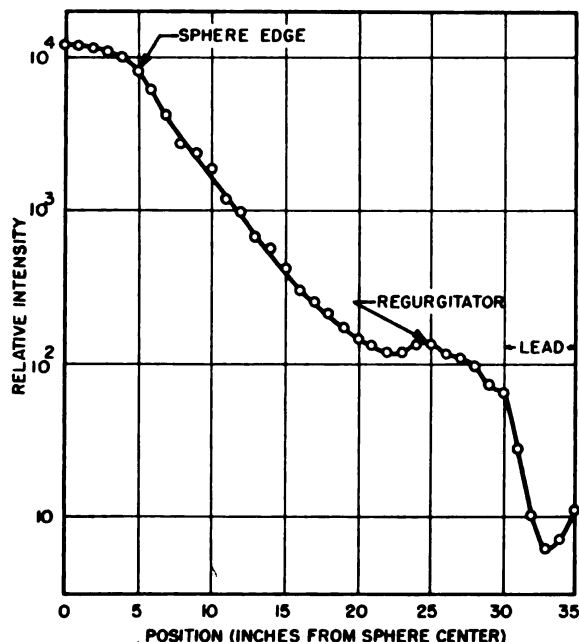


Fig. 1-84 Gamma-ray distribution in core and reflector at shutdown.

yielded a high background for the instrument, probably due to induced activity in the components. Short exposures had no measurable effect. The instrument was calibrated with a 300-mc Co^{60} source at the lowest range (<10 r/hr), and the slope was checked, by exposure adjacent to the sphere at higher power levels and at the edge of the reflector at various power levels. The slope was found consistent, by the latter technique, to the equivalent of 4×10^4 r/hr. Extension of the calibration by exposure in the sphere proved the slope consistent to 1.5×10^5 r/hr. The latter value was obtained in the center of the sphere at 500 watts immediately upon reaching that power level and is probably reliable within a factor of 2.

Gamma-ray traverse of sphere and reflector at shutdown. The detector described above was used to determine the gamma-ray distribution from the center of the sphere to the outer edge of the lead shield surrounding the reflector. The data are presented in Fig. 1-84. The data were taken 2 hr after shutdown to allow the glory-hole tube activity to decay.

Danger coefficients. Gross danger coefficients were taken on a variety of samples placed within the glory hole. Reactivity values assigned to the materials were determined by period measurements. In each case periods were taken periodi-

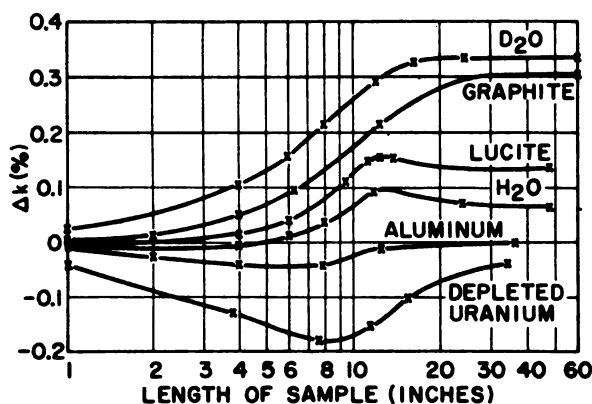


Fig. 1-85 Danger coefficients.

cally, with no sample in the glory hole, to determine the boiler reactivity and to make the minor corrections necessitated by fuel-temperature change. To minimize these corrections no cooling water or sweep gas was circulated through the sphere. A set of three periods at different power ranges was taken in each measurement. Periods were converted to Δk values by the water-boiler inhour formula.

The samples of graphite, aluminum, and Lucite used were 1 in. in diameter. All samples were geometrically centered in the sphere. The data are presented in Fig. 1-85. In the cases of liquid samples, the containers were thin-walled aluminum tubes. Danger coefficients were taken on the empty tubes, and these values have been added to those obtained with the tubes filled; the corrected data are presented. The tubes had an inner diameter of 0.94 in. and were completely filled. The D_2O was analyzed by the nuclear magnetic-resonance technique and found to contain 2.2 per cent

H₂O. The graphite is the same as that used in the reflector. The uranium used was depleted in the U²³⁵ isotope to approximately 60 per cent of its natural content. It was used in the form of solid metal cylinders 0.92 in. in diameter. The aluminum was type 3S.

In order to measure the period with the longer D₂O samples in the glory hole, the cooling-water coil was drained. This decreased the reactivity sufficiently to allow accurate period measurements.

Effect of loss of cooling water. In order to evaluate some of the danger-coefficient data and determine a possible, although remote, source of hazard, the cooling water was removed from the coil in the sphere. The volume of water contained in the cooling-line segment within the sphere is approximately 125 cm³. By period measurements it was determined that removal of the cooling water decreased the reactivity of the boiler 0.280 per cent. The reactivity contribution of the cooling water is, in this case, very nearly that of an equal volume of fuel solution.

Absorber traverse of glory hole. To obtain the effect of an absorber placed in the glory hole as

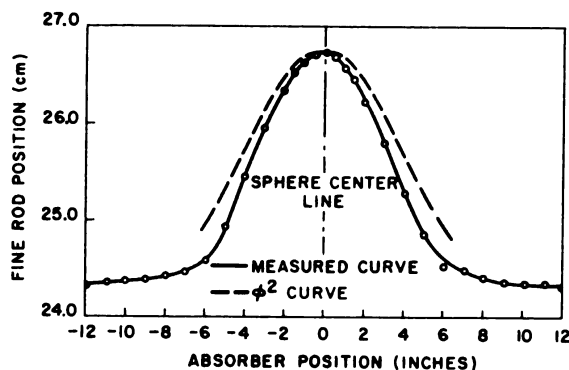


Fig. 1-86 Absorber traverse.

a function of its position, a small sample of boron was passed through the sphere in steps, and the critical settings of the fine-control rod were noted. The sample was composed of 20 mg of boron spread (nonuniformly) upon, and attached to, the surface of a 1/4- by 1- by 1/8-in. Lucite strip. This sample was wedged in the center of a 6-ft-long thin-walled aluminum tube. Figure 1-86 is a graphical representation of the data gathered. The function approaches that of the square of the

flux, and the difference between the two is probably due to the presence of the glory hole itself.

Boron effectiveness in glory hole. In an attempt to correlate the theoretical prediction of the effective pile cross section of the reactor with an experimental value, boron, in increasing quantities, was added to the sphere and the change in k noted. The experimental result was about 35 per cent lower than the predicted value. The discrepancy is probably due to the difference in effectiveness between neutrons found at any point in the glory hole and those found at corresponding locations elsewhere in the core.

Five samples of boron powder were made and tested. The boron was brushed uniformly on scotch tape and the net weight of boron determined. The samples were 4 in. long by 1 in. wide. Approximately 7 mg was contained on each sample, giving a layer of boron somewhat less than 0.3 mg/cm².

The results of the experiment are tabulated in Table 1-13.

Table 1-13

Total B in sphere, mg	$-\Delta k$, %	$-\Delta k$, %/g B
8.02	0.028	3.49
15.64	0.055	3.52
22.44	0.078	3.48
29.25	0.101	3.45
36.25	0.124	3.42

Control-rod temperature coefficient. In order to ensure that the fine-control rod did not change in effectiveness as the solution level rose (because of temperature increase or the solution density being lowered), an experiment was conducted to measure such an effect. This point has importance in normal operation and particularly in hazard considerations.

It was theorized that if the solution expanded, filling the sphere volume which is normally vacant, the improved geometry might result in a shift of the flux center of gravity and a subsequent decrease in the effect of the fine-control rod and No. 2 safety rod, both of which are underneath the sphere. Solution expansion itself gives a marked decrease in reactivity, and therefore the over-all

effect, considering improved geometry vs. decreased density, would be a loss in reactivity. However, it is desirable to know if the control and safety devices undergo any change in effectiveness under such circumstances and, if so, what minimum reactivity they may be expected to compensate under the most adverse conditions.

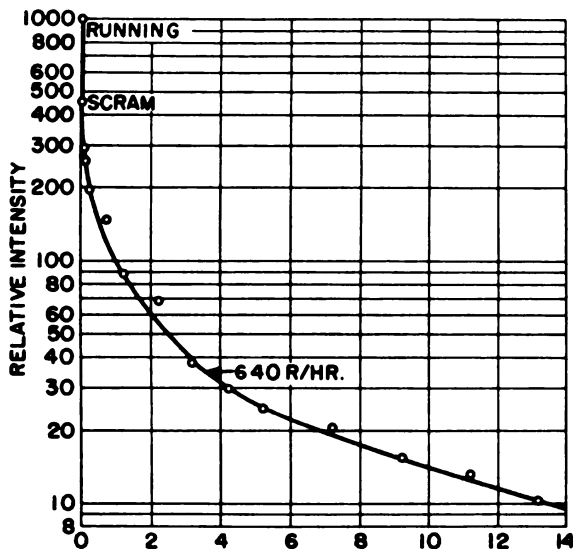
The effect was determined in the range of 60 to 88.5°F solution temperature. The solution was heated in 10° steps and the critical position of the fine-control rod noted. In order that the critical position not be changed appreciably, the reactor was poisoned with depleted uranium at the 60°F point and enriched with Lucite and D₂O in the glory hole at the 80° and 88.5°F points. The reactivity values of these samples had been tested previously and were well known. Since the temperature coefficient was also known, an accurate estimate of the net reactivity change that the control rod was required to compensate was possible. To illustrate, if the boiler is critical at 70°F and is heated to 80°F as 10 in. of Lucite is centered in the glory hole, the net reactivity change is only -0.024 per cent k . If the critical position of the fine rod is found to be nearer the ALL-IN position than is required to compensate the net reactivity change, it is said to have overcompensated. Conversely, if the new critical setting for the rod was found to be farther out than expected, it is said to have undercompensated. The data are presented in Table 1-14.

Table 1-14

Temperature, °F	Undercompensation, % Δk	Overcompensation, % Δk
60		
70	0.037	
80	0.024
88.3	0.007

The results do not lend themselves to ready error assignment. However, a pessimistic estimate would be ± 0.011 per cent Δk , which indicates the effect to be present, although not consistent in direction. In the range measured, it is definitely not of significance with regard to hazard considerations.

Decay of fuel-solution gamma activity. After running at a power level of 500 watts for approximately 3 hr, the reactor was shut down. The residual gamma-ray activity was recorded for a period of slightly more than 13 hr. The detector was placed in the horizontal thermal-column beam hole within 1 to 2 in. of the sphere. Its location placed it just outside the No. 1 safety rod. By line of sight, therefore, it was shielded



from the solution by approximately 0.15 in. of steel, by the boron-carbide packing within the safety rod, and by approximately $\frac{1}{2}$ in. of graphite. The decay is plotted in Fig. 1-87.

Pressure coefficient of reactivity. The static pressure above the fuel solution was adjusted between 0 and 60 in. water in order to measure the effect on reactivity of the boiler. The reactor was critical at low power with no sweep-gas circulation when the measurements were performed. The change in the critical position of the fine-control rod was transcribed to reactivity terms by use of the control-rod calibration curve.

The results obtained in this experiment are unique to this reactor since the fill and drain line to the sphere is valved-off from the atmosphere. The effect of a decrease in pressure above the fuel solution is, therefore, a function of the volume of air trapped in the dead end of the filling line. If this volume were zero, for instance, there would

be no reactivity change expected since the surface level of the fuel solution in the sphere would not change with variation in pressure. The coefficient is caused by solution being drawn into the sphere from the fill line as the system pressure is decreased.

Figure 1-88 presents the data obtained. The effect presents a small but measurable error if danger-coefficient measurements are made with the gas system in operation. The pressure fluctuations allowed by the automatic control system would be equal to ± 0.0018 per cent Δk .

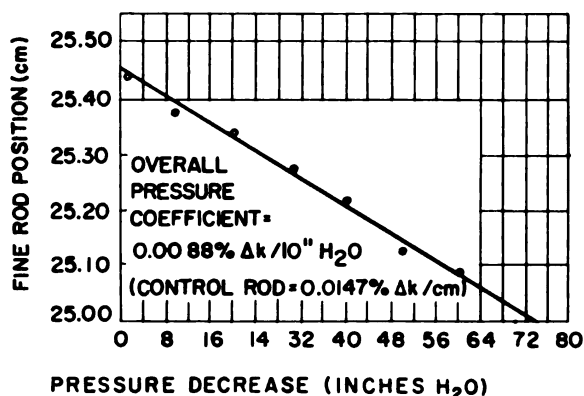


Fig. 1-88 Pressure coefficient of reactivity.

Fine-control-rod calibration. The fine-control rod has been calibrated in terms of equivalent reactivity versus position.

The calibration of the control rod was arrived at by successive period measurements. This was possible only for that portion of the control-rod travel representing normally supercritical positions. The measurement was extended to the ALL-IN position by inserting D₂O in the glory hole and cooling the fuel solution to 60°F, 10° below the normal danger-coefficient measurement temperature. From the results of the measurement it may be concluded that the fine-control rod is capable of controlling 0.669 per cent Δk . The calibration curve of the control rod is given in Fig. 1-89.

Effects of reflector voids. Loss of reactivity resulting from the removal of certain pieces of the graphite reflector has been measured.

The beam-hole plug, which is 2.0 in. in diameter and leaves a void whose center line is in the horizontal equatorial plane of the sphere, causes a

loss of reactivity of 0.046 per cent upon removal.

The effects of individual stringers which can be removed without causing the boiler to be subcritical with both control rods out have been determined. When withdrawn, each stringer leaves

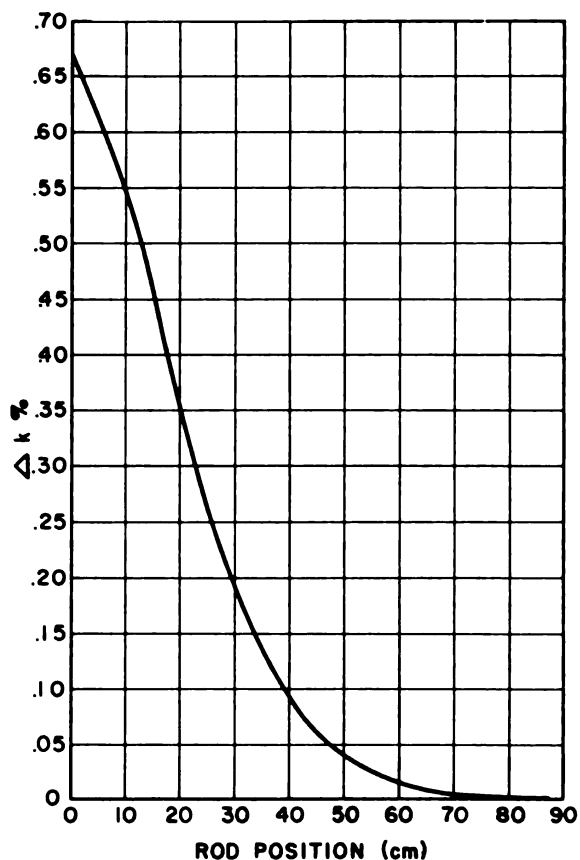


Fig. 1-89 Fine-control-rod calibration.

a void in the graphite 4 in. square and approximately 36 in. long. Table 1-15 is a tabulation of the data collected and includes the minimum radial distance from the center of the sphere to

Table 1-15

Stringer number	Distance, in.	Reactivity loss, % Δk
1	16.1	0.058
4	14.5	0.086
5	16.1	0.065

the center lines of the voids. The discrepancy between stringers 1 and 5 is unexplained.

Figure 1-90 is a graph showing the effectiveness of filling the void caused by removal of stringer 6.

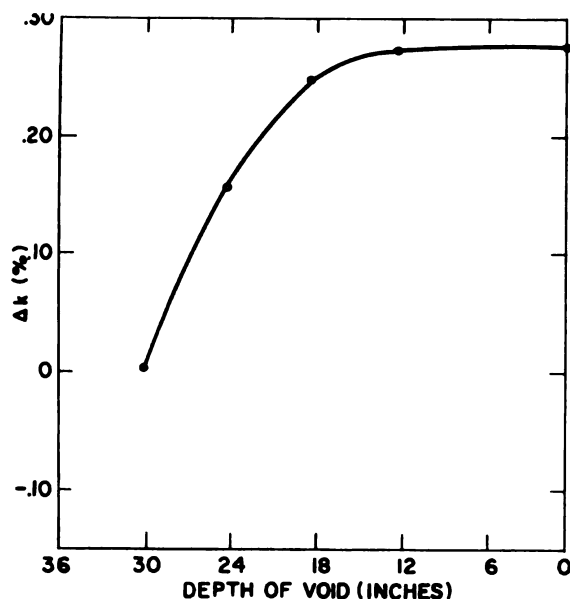


Fig. 1-90 Decrease of void-depth effect on reactivity.

A continuation of these measurements consisted of reducing the void to a $1\frac{1}{4}$ -in.-square cross section with its center line $1\frac{1}{4}$ in. above that of the stringers. These voids were the same length as the stringers and were made possible by removal of a graphite foil holder from a specially fabricated stringer. Table 1-16 gives the effects of

Table 1-16

Void (stringer number)	Distance, in.	Reactivity loss, % Δk
1	17.2	0.004
5	17.2	0.004
4	13.6	0.014
6	10.1	0.035
2	10.1	0.052
3	8.9	0.079
7	8.9	0.076

these voids and their minimum center-line distances from the sphere center.

The discrepancy between voids 6 and 2, which are equidistant from the core, is undoubtedly due to the spillover tank-sphere pipe, which crosses at a right angle to and within an inch of void 2. Figure 1-91 presents these data graphically.

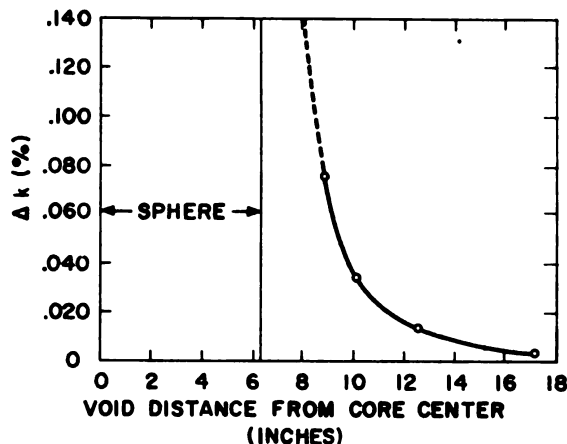


Fig. 1-91 $1\frac{1}{4}$ -in.-square void effects.

Response time of safety devices. An experiment was performed to determine the length of time, under normal operating conditions, that is required for the safety rods to be seated after a scram-level flux is presented to one of the ion-chamber power detectors.

The reactor is so interlocked that it will shut down automatically and/or cannot be started up for any of the following reasons:

1. Excessive power level.
2. Excessive power-increase rate.
3. Excessive sweep-gas pressure.
4. Excessive fuel temperature.
5. Excessive hydrogen concentration in sweep gas.
6. Earthquake.
7. Low gas-flow rate.
8. Water leak in gas-pressure regulating system.
9. Operator action.
10. Instrument power lost.
11. Startup source withdrawn from reflector.
12. Biological shield doors open.
13. Safety switches open inside biological shield.

The only automatic safety circuit listed which may be evaluated with reasonable ease and meaningfulness in terms of time response is the exces-

sive power level. This circuit consists of a gamma-compensated slow-neutron ionization chamber, the signal from which is amplified by a micromicroammeter and transmitted to a sensitrol relay. Approximately 90 ft of coaxial cable (13

sec. The withdrawal time of the cadmium shutter placed between source and chamber is unknown since it was done manually. The shutter was light in weight and was required to move only 4 to 5 in. so a reasonable withdrawal time might be

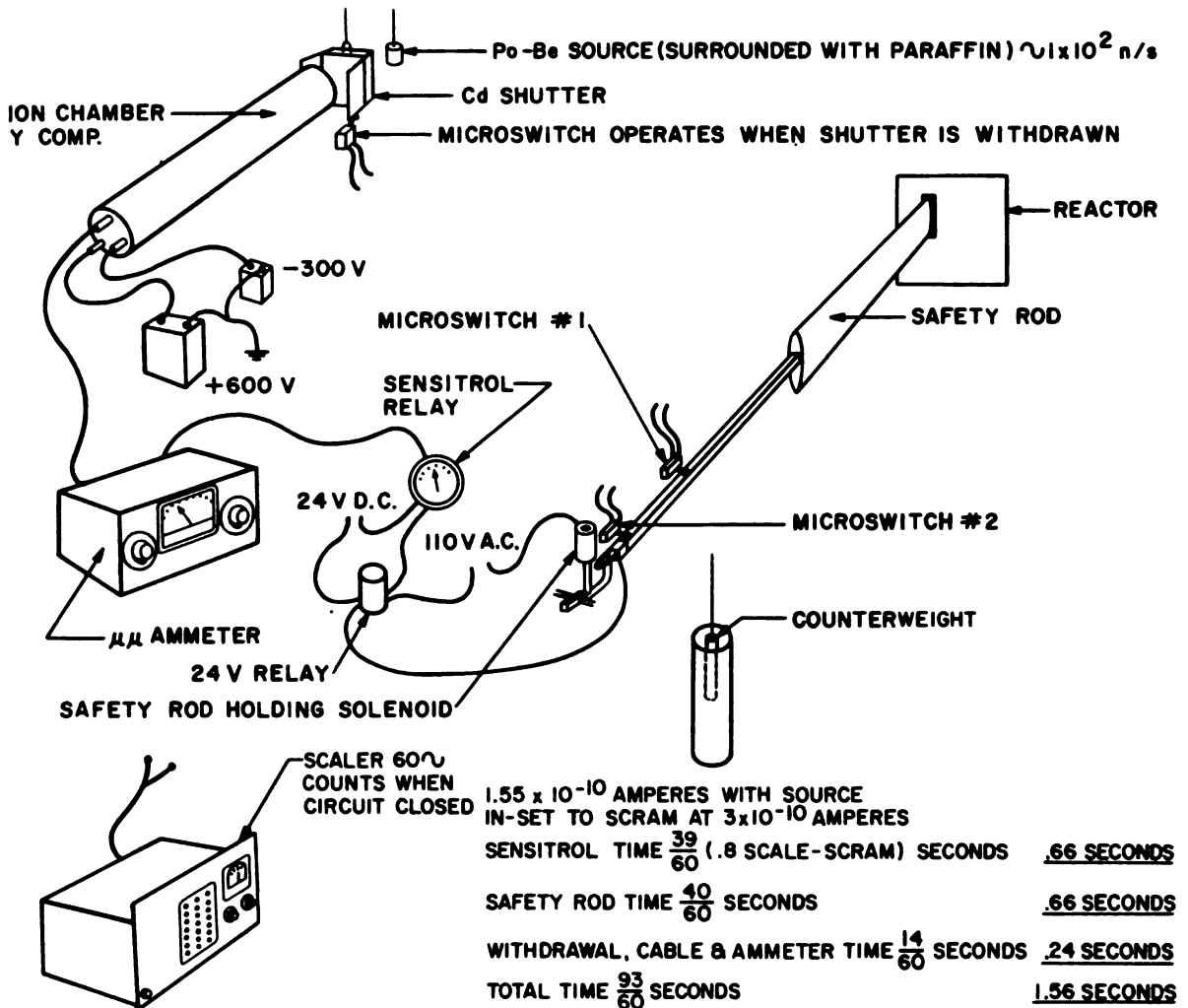


Fig. 1-92 Safety system.

$\mu\mu$ /ft) connect the ion chamber and micromicroammeter. At the closing of this relay, a second relay is opened, which interrupts the circuit of the holding solenoid for the safety-rod mechanisms. The safety rods travel a total of 3 ft and are accelerated by lead counterweights.

The general arrangement for the experiment is shown in Fig. 1-92. There are two obvious factors which limit the accuracy of the final value of 1.56

assumed as no greater than 0.05 sec. The statistical error involved in the timing mechanism is ± 0.017 sec. The closing and opening times of auxiliary microswitches and relays used are not subtracted from the elapsed times, since in most cases these values cancel themselves and in any case are small.

The micromicroammeter received a constant signal of half scale on its 3×10^{-10} -amp range

before the step increase caused by shutter movement was applied. The higher signal was of the magnitude required to trip the circuit.

The sensitive relay was found to close in 0.65 sec when tested alone and full-scale current was applied. This time increases as much as 40 per cent in moving the trip level from 80 to 90 per cent full scale.

The rod travel time and holding-latch opening time were measured compositely as 0.66 sec.

By difference between the total time of the system, 1.56 sec, and the times quoted above, the ion-chamber response time was determined as 0.19 sec. The same ion chamber connected to each of two identical circuits gave response times differing by only 7 per cent. Repetitive trials on the individual system resulted in maximum deviations of 2 per cent.

Absorber orientation effect in glory hole. In testing for the effectiveness of boron in the glory hole, it was observed that a disk of absorber material exhibited a larger poisoning effect on the reactor if it were placed in the horizontal rather than the vertical plane. Gold foils $\frac{1}{2}$ in. in diameter, used to determine neutron distribution in the core, give the same result regardless of their orientation. This indicates that the neutrons intercepted by a vertical absorber in a given position have a lower utilization factor in the boiler than those neutrons intercepted by a horizontal absorber and that the total neutrons intercepted by the absorbers are the same in number.

Shutdown power level. The shutdown power level of the reactor has been measured by standardized foil activation in the center of the sphere. This power level was found to be 0.12 mw with safety and control rods fully inserted and the startup source (Po-Be, 10^6 neutrons/sec) 13.6 in. from the center of the sphere.

FIFTY-KILOWATT SOLUTION-TYPE REACTOR

General Design Features. The 50-kw solution-type reactor shown in Fig. 1-93 consists of a solution of uranyl sulfate, contained in a spherical tank, located at the center of a graphite cylinder. The graphite is contained in a cylindrical tank which provides some thermal-neutron shielding.

The entire reactor is enclosed in a shield of dense concrete.

As may be seen from the following discussion, the basic design features are the same as those previously described for the 1-watt and 50-watt reactors. The 50-kw reactor, which has been designated as the L-8, is planned for construction at the Armour Research Foundation in Chicago, Illinois. Figure 1-99 of the Appendix drawings shows additional details of this reactor.

Core and Fuel-handling System. The spherical-core tank is made by welding two stainless-steel hemispheres together. The sphere has an inside diameter of $12\frac{1}{2}$ in. and is filled with approximately 14 liters of uranyl sulfate solution containing approximately 800 g of highly enriched U^{235} .

Four control-safety-rod thimbles and a fill-and-drain line pierce the upper half of the spherical shell. Three parallel cooling coils, two gas lines, and a liquid-return line extend down through the lower half of the sphere and into the gas-handling and heat-exchanger room below. A $1\frac{1}{2}$ -in. entrant tube enters the core obliquely from above.

A thin-walled liquid-tight outer envelope of aluminum is placed around the sphere and its associated piping. In case of a leak, most of the uranyl sulfate solution could be withdrawn into the fill-and-drain tank, all the coolant and gas-handling tubes pinched off and disconnected, and the entire core unit safely removed. None of the reflector graphite or surrounding atmosphere would become contaminated, and a new core unit could easily be installed.

A fuel-solution overflow is incorporated in the gas-handling system, providing an additional safety feature. An uncontrolled power burst would cause a very rapid formation of gas in the core solution, forcing much of the solution out of the core into a spillover tank and causing the core to become subcritical. The geometry of the core is such that as soon as the liquid level starts to rise, the reactivity starts to decrease; the fuel solution is trapped in the spillover tank and slowly pumped back into the core tank by the condensate return pump.

Cooling System. The reactor core and certain gas-handling components are cooled by a closed

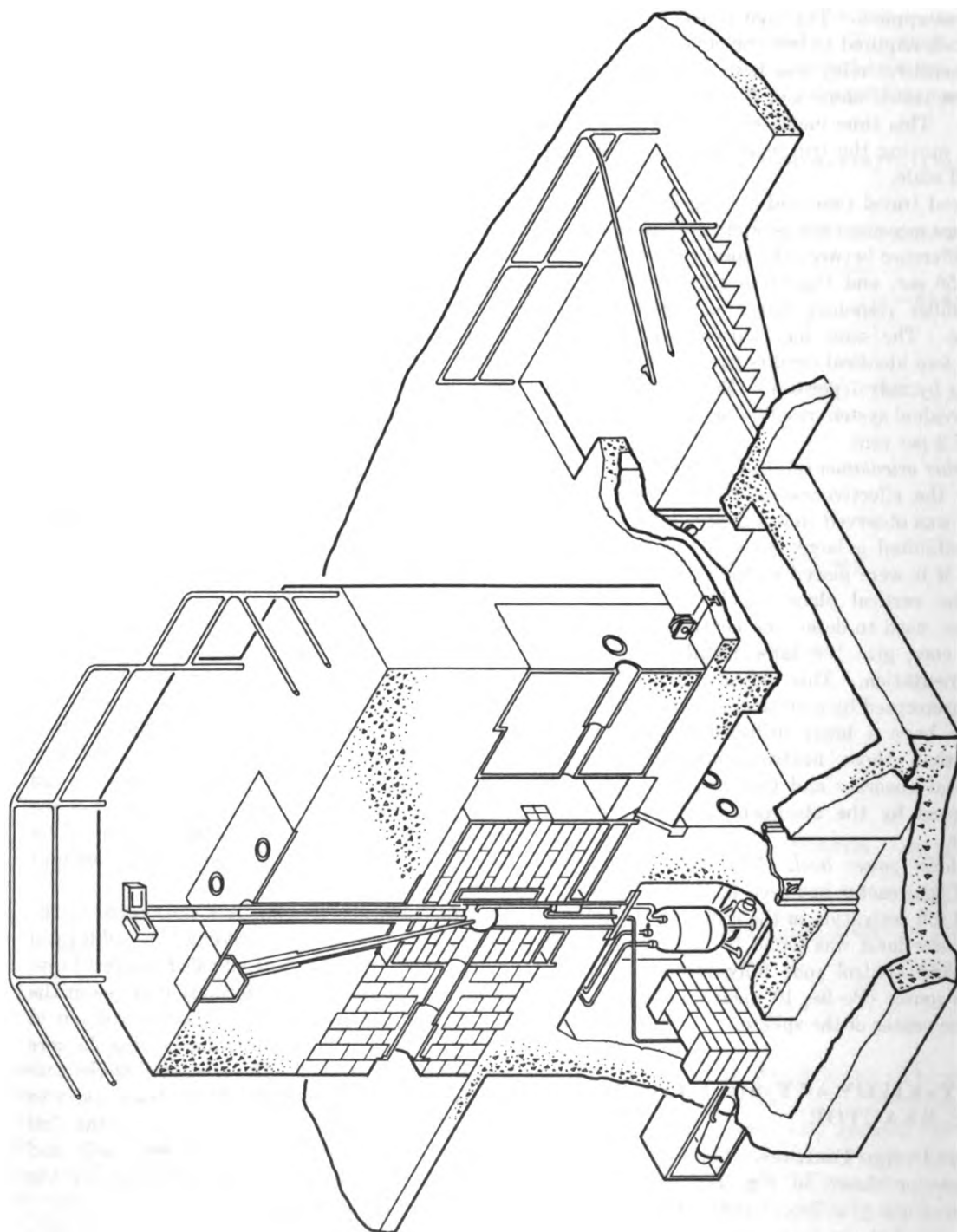


Fig. 1-93 Solution-type research reactor, isometric.

recirculating refrigerated system using distilled or demineralized light water. The core-cooling loop consists of (1) approximately 60 ft of stainless-steel tubing in the core; (2) the associated pump, valves, and piping; and (3) a water-to-Freon heat exchanger. The heat exchanger is a standard water chiller of the type used in air-conditioning systems.

Thermocouples indicate the temperature of the core and of the coolant. The electrical conductivity and the level of the water in the sump are measured to indicate contamination or leakage.

A motor-operated valve is mounted in the core-coolant supply line so that the flow through the core coils can be changed to meet various power-level requirements. The position of the valve is indicated at the control console. A flow switch in the coolant line indicates any interruption of coolant flow.

The coolant pump, instrumentation, and water-to-Freon heat exchanger are located in the same shielded room as the gas-handling system. For accessibility and service, the Freon condenser is located outside the shielded room.

Gas-handling System. The gas-handling system performs the following functions: (1) It contains the fission gases in an entirely closed system, (2) it recombines the radiolytically decomposed water, and (3) it provides a means of periodic removal of fission-product gases. The hydrogen and oxygen evolved from the radiolysis of the water in the core are diluted with a sweep gas to form a nonexplosive mixture. This mixture is then recombined catalytically over platinized alumina. The dilutant gas is circulated by a positive-displacement blower. A lift pump is used to return the condensate to the core. The entire system is designed to be gastight and explosion-proof.

Reflector and Shielding. The reflector around the core consists of graphite stacked in a cylindrical steel tank 5 ft in diameter and 5 ft high. Appropriate holes for beam, irradiation, and instrumentation tubes are located in the graphite. The reflector section above the core is placed so that it can be installed or removed with the core. All the graphite is supported and held as a unit

by the tank, and the tank is firmly anchored to the foundation.

The shielding around the reflector tank consists of approximately 5 ft of dense concrete. The shield is poured monolithically on two sides and the top of the tank, forming a U-shaped shield. The other two sides are closed by large, removable concrete blocks. This permits the addition of a second thermal column or other irradiation facility subsequent to the initial installation. In addition, it provides good working access while the reactor is being built. The thermal column and beam holes are steel lined. The thermal column is shielded by a rail-mounted concrete-filled steel door. The smaller experimental holes are closed by laminated iron-masonite plugs.

Control System. Four vertically oriented rods provide regulation and shutdown for the reactor. These rods are located in thimbles which pierce the core tank; three are used for regulation, and all four function as shutdown rods. All four rods are automatically released, and fall by gravity when the reactor is scrammed. The positions of the rods are indicated on the control console by means of a selsyn system.

The four rods which pierce the core control a total of about 8 per cent excess reactivity. Location of the rods inside the core tends to minimize "shadowing" of the experimental facilities.

Instrumentation. Adequate control of the reactor is provided by two types of sensing elements which provide neutron-level information over the range of 0.1 mw to 50 kw. In the power range up to approximately 1 watt, a fission chamber is employed. The pulses from this unit are amplified and fed into a counting-rate meter which produces a voltage proportional to the number of pulses received per unit time. This voltage is fed to a recording potentiometer which is calibrated directly in power. To prevent damage to the fission chamber, it is withdrawn from the vicinity of the core before its upper counting limit is reached.

In the power range from approximately 0.5 watt to 50 kw, three gamma-compensated ionization chambers are employed with a 6-decade range. The currents (10^{-10} to 10^{-4} amp) of two of these chambers are fed into electrometers, which in turn provide a signal to the level recorder.

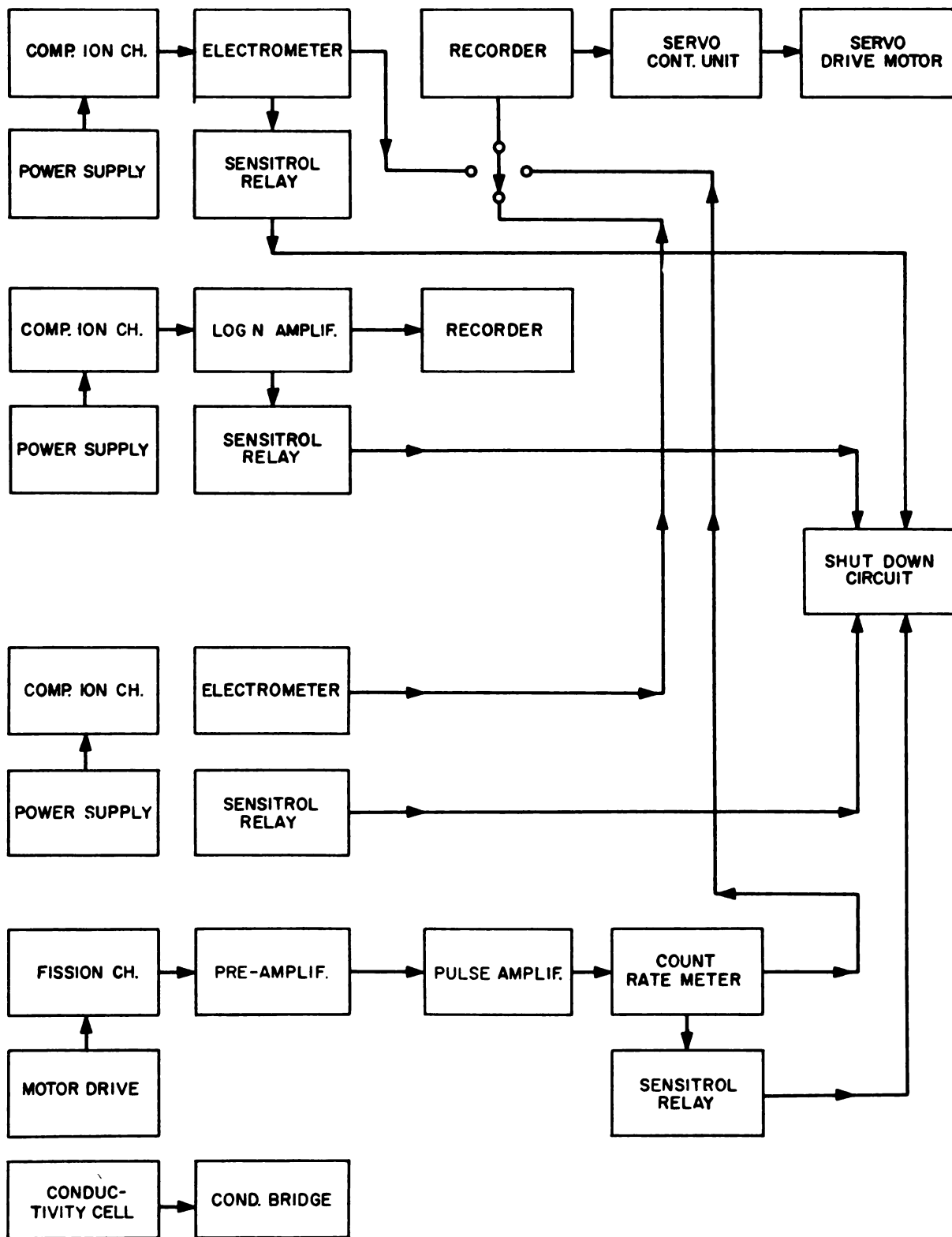


Fig. 1-94 Block diagram of instrumentation system.

The power level is automatically maintained by feeding an error signal from the level recorder to a servoamplifier, which drives the regulating-rod motor in the proper direction to correct the error. The signal feeding the automatic control system can originate from either of two ion chambers or the fission counter tube. Provision is also made for manual control of the regulating rods.

The third ionization chamber feeds a logarithmic amplifier. This unit provides two outputs. One output is a voltage proportional to the logarithm of the neutron flux. The second output is a voltage proportional to the time derivative of the log flux and is thus a measure of the inverse period of the reactor. The log flux signal is fed to a recording potentiometer. Both signals are displayed on meters.

The over-all response time of the two high-level channels from ionization chamber through the rod-holding solenoids is ≤ 0.5 sec. This response time is considered adequate to provide safe shut-down for ordinary power surges. In the event of a very short period increase in power, the fuel solution will be forced into the spill chamber and the reactivity will drop.

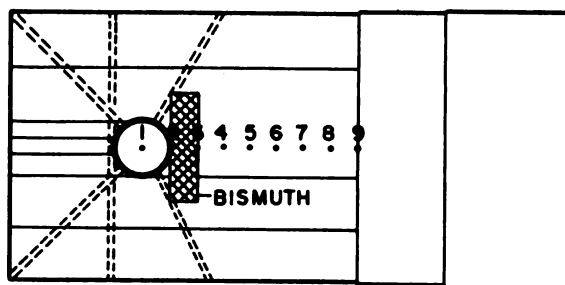
Table 1-17

Quantity	Type
1 beam tube.....	6-in. diameter extending radially to within 2 ft of the core tank and 4-in. diameter from there to the core tank
2 beam tubes.....	4-in. diameter extending radially to the core tank
2 beam tubes.....	3-in. diameter extending radially to the core tank
2 through tubes...	2-in. diameter passing tangentially to the core tank
1 entrant tube....	1½-in. diameter passing through the central region of the core
4 vertical tubes....	4-in. diameter located in the reflector, 2 ft from the core tank
1 thermal column..	5 ft square with a 12-in.-square central region which can be removed
1 gamma facility..	The room beneath the reactor, which houses the gas-handling system, arranged for gamma irradiations from the 50,000 curies of radioactive gas present at 50 kw

The reactor is shut down automatically in the event the flux level or the period exceeds preset levels or a serious malfunction occurs in the auxiliary equipment. The shutdown is accomplished by deenergizing the magnets which hold the con-

trol and safety rods, causing them to fall under gravity into the core. Sensitrol relays are used as the basic flux-level and period-trip elements. A block diagram of the instrumentation is shown in Fig. 1-94.

The operation of the reactor gas-handling system is observed by measuring temperatures, pressure, and flow rate. The gas pressure is measured



APPROX. FLUX AT 50 KW.		
POINT	FAST NEUTRONS CM ² /SEC	THERMAL NEUTRONS CM ² /SEC
1	1.6×10^{12}	1.7×10^{12}
2	4.2×10^{11}	9.3×10^{11}
3	2.0×10^{10}	7.5×10^{11}
4	3.2×10^9	5.8×10^{11}
5	6.5×10^8	4.3×10^{11}
6	1.5×10^8	2.6×10^{11}
7	3.6×10^7	1.3×10^{11}
8	9.8×10^6	5.8×10^{10}
9	2.6×10^6	5.6×10^9

GAMMA FLUX AT POINT 9 400 REP/HR
Fig. 1-95 Flux-distribution map.

with a strain-gauge-type pressure cell, the flow rate with an electromagnetic sensing element. Core temperatures are also measured. The induced activity of the cooling water is monitored with a conductivity cell.

Experimental Facilities. The arrangement of exposure facilities is shown in Fig. 1-93. These facilities are summarized in Table 1-17.

The beam tubes are used for internal reactor exposures as well as for beam experiments outside the reactor. The tangential through tubes are ideally suited for the installation of apparatus of the "pneumatic-rabbit" type for exposure of short-lived isotopes. The reentrant tube passing

through the central region of the core permits access to the region of highest neutron flux. The vertical holes are for extended isotope irradiations and could be used for the installation of selective multiple-specimen irradiation facilities or of stringer-type irradiation facilities. The thermal column, with its removable sections, permits the exposure of various materials to neutron fluxes of varying energy distributions. The gamma-irradiation facility is suited for biological and chemical experiments where a large volume of gamma rays of relatively low intensity is required. Gamma radiation of about 50,000 curies is available from the radioactive xenon and krypton fission gases circulating in the gas-handling system at a reactor power of 50 kw.

Operating Characteristics. In this highly enriched, light-water-moderated reactor, a peak

thermal flux of approximately 1.7×10^{12} neutrons/cm²/sec and a peak fast flux (≥ 0.1 Mev) of about 1.6×10^{12} neutrons/cm²/sec are produced at the maximum design power of 50 kw.

The distribution of the neutron flux as a function of distance from the center of the core is indicated in Fig. 1-95 at a reactor power of 50 kw. The gamma flux at one location in the thermal column is also shown in Fig. 1-95.

The use of a large 5-ft-square thermal column and a bismuth, instead of lead, gamma shield provides a maximum thermal-neutron beam with a minimum number of fast neutrons and gamma rays.

The gamma flux reported is almost entirely due to thermal-neutron absorption in the graphite. Hence, raising the thermal-neutron flux produces a corresponding increase in the gamma flux.

APPENDIX

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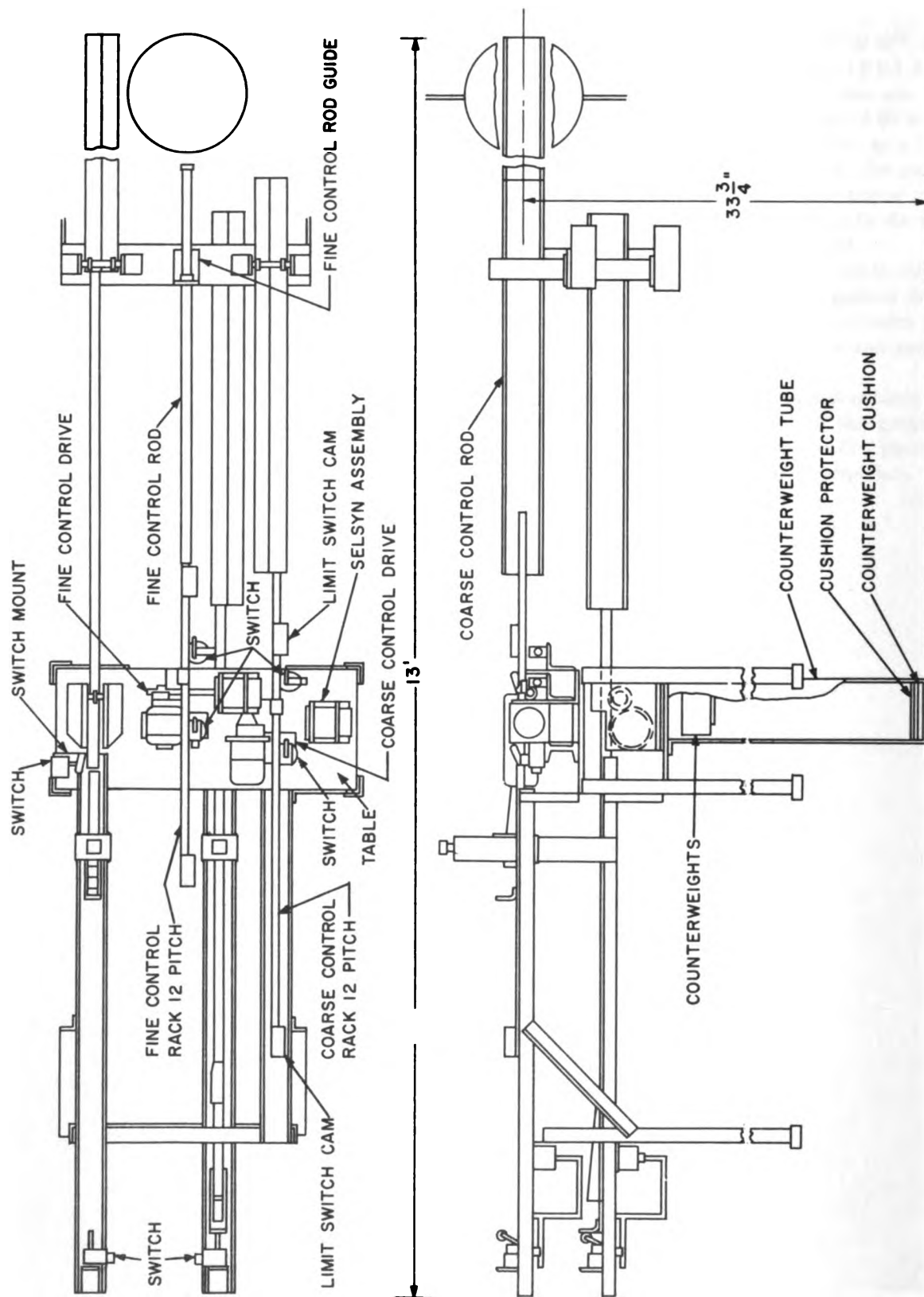
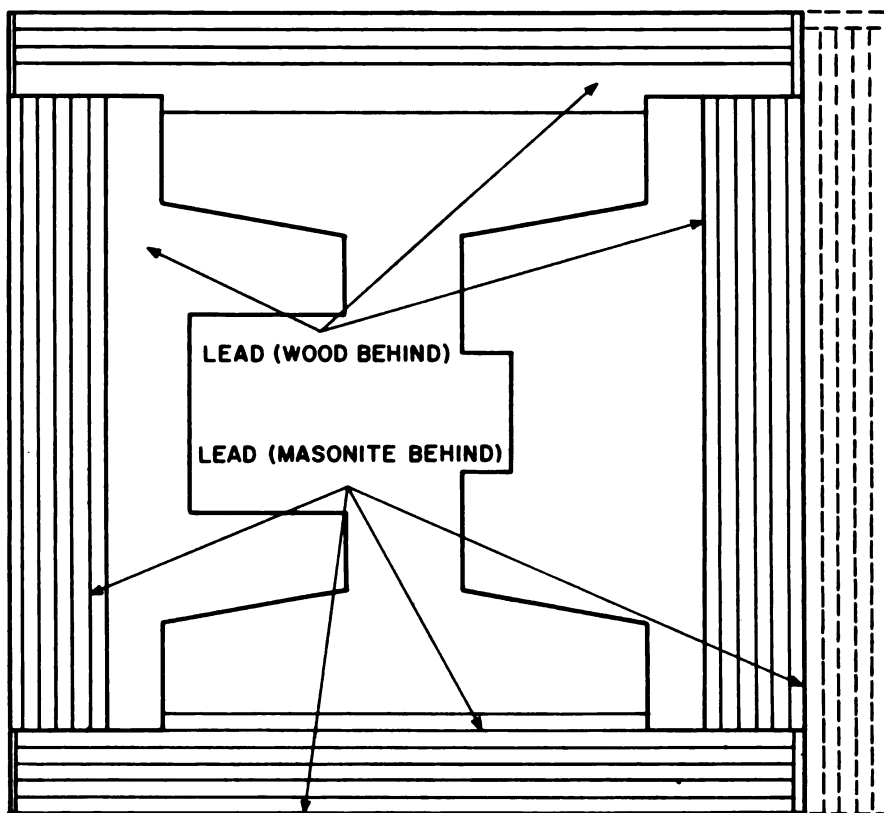


Fig. 1-96 Rod assembly, 1-watt WBNS.



DETAIL A

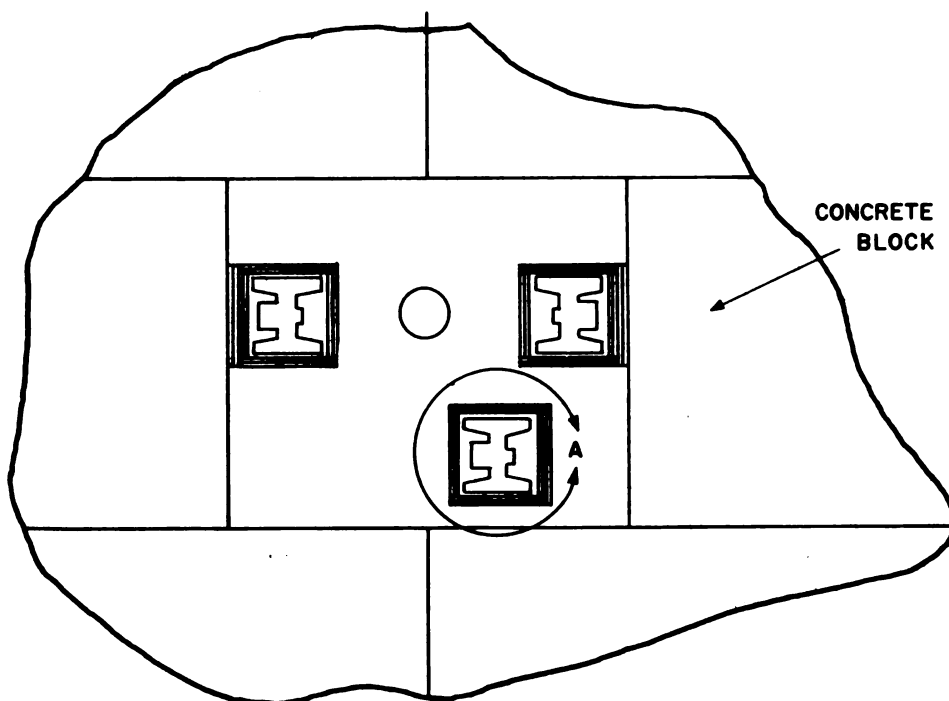


Fig. 1-97 Plug assembly, 1-watt WBNS.

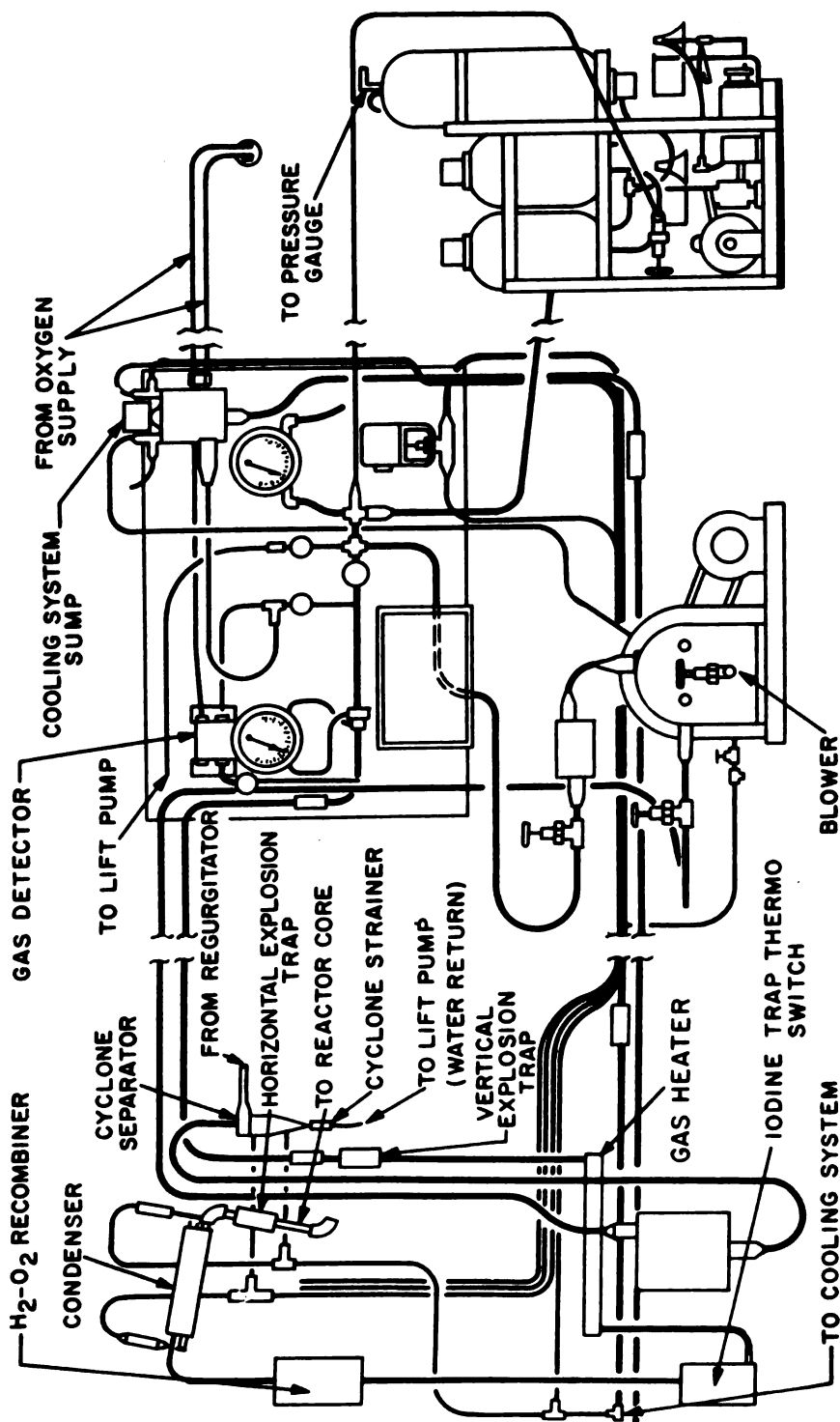


Fig. 1-98 Closed-cycle gas system, 500-watt.

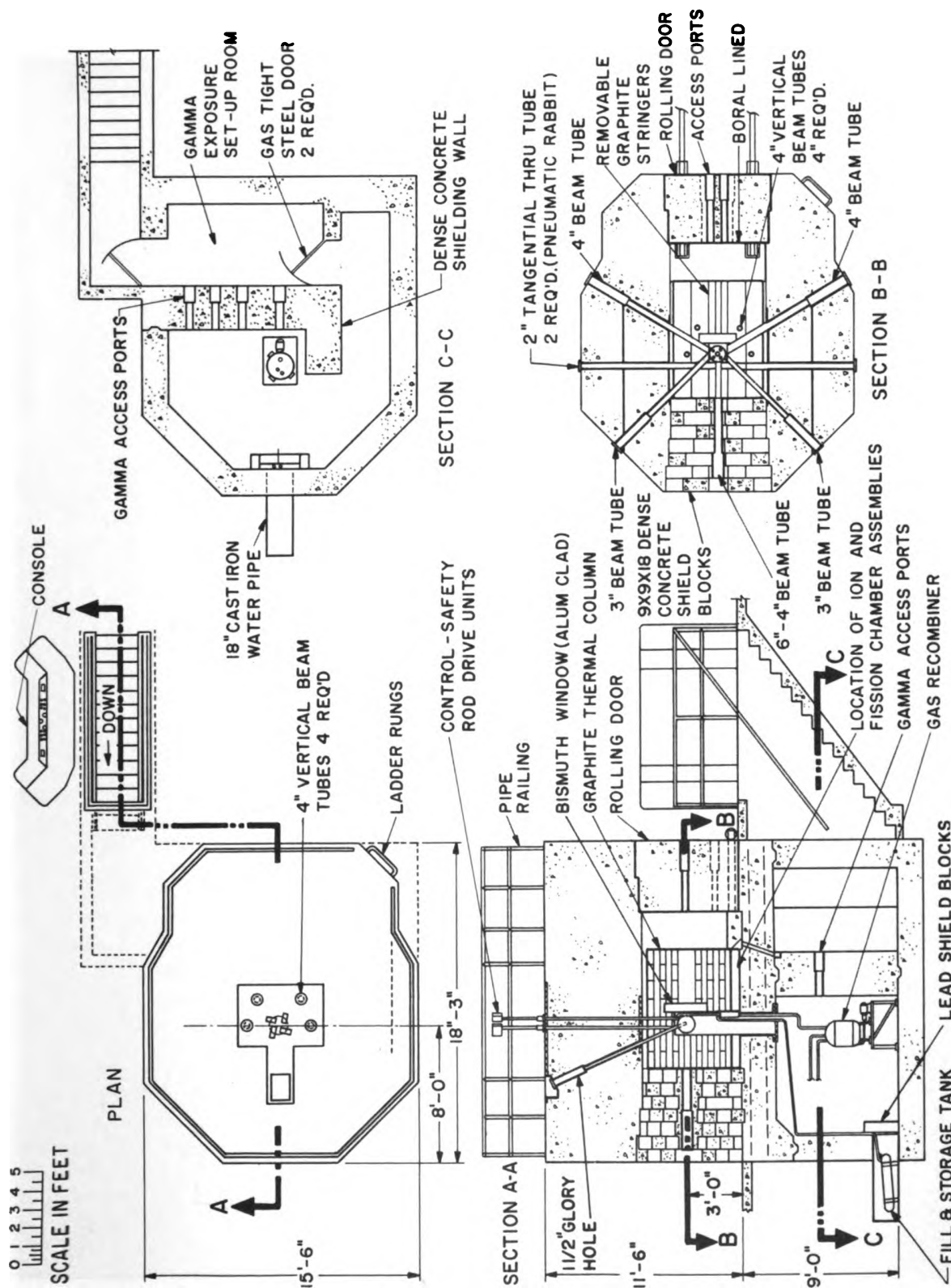


Fig. 1-99 Solution-type research reactor layout, 50-kw.

CHAPTER 2

Light-water-moderated Reactor

TYPE II *Heterogeneous—Enriched Fuel*

COMPILED BY OAK RIDGE NATIONAL LABORATORY
CARBIDE AND CARBON CHEMICALS COMPANY

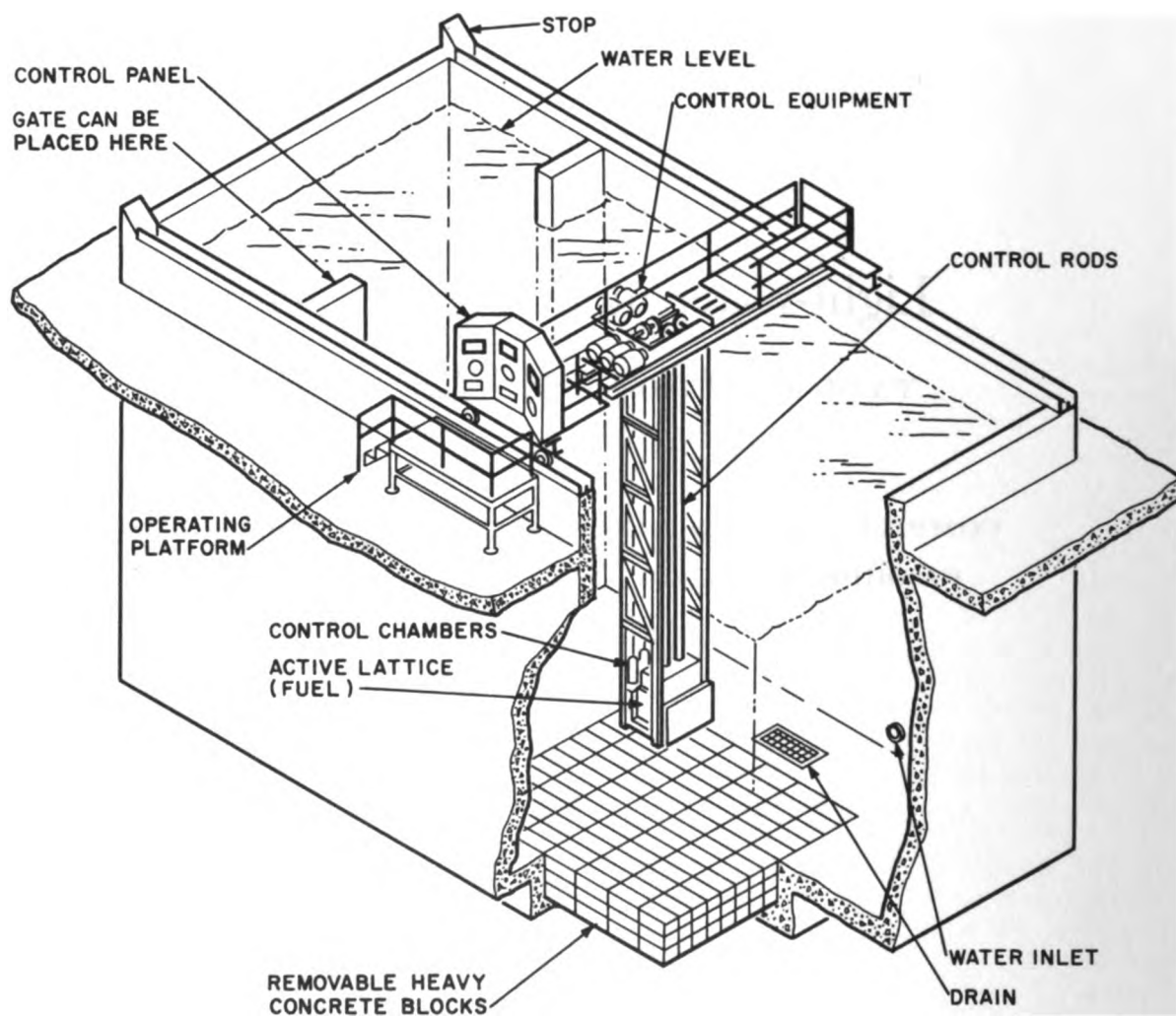


Fig. 2-1 Sketch of bulk-shielding reactor (BSR) operating at the Oak Ridge National Laboratory. This light-water-moderated pool reactor is of the heterogeneous enriched-fuel type and provides high thermal-neutron flux, accessibility, and versatility at low cost.

CHAPTER 2

Light-water-moderated Reactor

TYPE II *Heterogeneous—Enriched Fuel*

Light-water-moderated heterogeneous reactors make ingenious use of ordinary water as a moderator and, sometimes, also as a coolant and a shield. The design is remarkably simple, consisting essentially of a lattice of fuel elements immersed in a large pool of water, as apparent from Fig. 2-1. The term *pool reactor* is popularly given to this type of construction. The chief advantages are low cost, safety, flexibility, and accessibility.

The prototype of this class of reactors is the bulk-shielding reactor (BSR) at the Oak Ridge National Laboratory in Oak Ridge, Tennessee, which was first placed into operation in December, 1950. The bulk-shielding reactor, in turn, resulted from the development of the materials-testing reactor (MTR), now operating in Arco, Idaho (see Type III). The operation of the bulk-shielding reactor has proved quite successful, and a number of copies with relatively minor modification are being built, engineered, or planned. This type of reactor may be considered representative of pool reactors in general, and consequently the following description will be confined to the principal design features of the bulk-shielding reactor. In addition to design information, sufficient operating and experimental data are presented to permit a conception of the scope of activity for which this reactor facility is applicable.

A more recent type of pool reactor, known as *aquarium*, is discussed immediately after the description of the bulk-shielding reactor. The *aquarium reactor*, which is available for inspec-

tion at Geneva, differs slightly from the BSR, but its essential operation is the same.

The high degree of accessibility of the pool-type reactors allows a wide choice of diversified research programs. Some examples may be cited. Interesting investigations on the effects of radiation on biological processes may be planned and performed. Radiation-damage work on large or small components can be carried out. Large components may be irradiated for tracer-type wear tests. The reactor may, of course, also be used for its original purpose of studying the attenuation of radiation in various materials.

Other types of research, not depending upon accessibility, may also be carried on. For example, a collimated neutron beam can be obtained easily by leading an empty pipe through the water to the reactor core. Such a beam will serve as a source of sufficient intensity for neutron-diffraction equipment, or it may be used in conjunction with velocity selectors.

Chemical analysis by means of radioactivation is another research possibility. The reactor will supply a neutron flux which is ample for satisfactory bombardment of samples. Radioactive isotopes with fairly intense specific activities can be produced, with the short-lived ones being of special interest. Specific types of experimental programs are discussed under Experimental Methods.

Engineering drawings of the major components are given in the Appendix C to this chapter. Complete detailed drawings of parts are omitted, however, to conserve space. Furthermore, it was

felt that a potential user would probably wish to make a number of changes which would invalidate such details.

DESCRIPTION OF FACILITY

Reactor. The laboratory building of the bulk-shielding facility contains both the reactor part of the facility and the areas used for supporting activities. The reactor part of the facility consists of the pool and water-handling equipment, the reactor and supporting structures, and the instruments used with the reactor.

The bulk-shielding reactor is essentially an assembly of enriched-uranium fuel elements in a pool of water. The fuel elements are hollow bundles of plates, through which water flows by natural convection. Each plate consists of an aluminum-uranium-aluminum sandwich. The water of the reactor pool circulates between the fuel plates, thus acting as moderator and coolant. In addition the pool water also serves as reflector and shield. Other reactor components include control rods, sensory instruments, and safety circuits.

The height of the active lattice is 24 in. A typical set of measurements for the other dimensions might be 15 by 18 in. However, the configuration for critical loading may be changed by merely rearranging the assembly of fuel elements. Thus reactors of varying configuration may be obtained, and the leakage flux may be peaked on any side of the reactor frame. The reactor assembly is movable in one direction along the reactor pool. Operation is quite flexible; the reactor could even be operated while supported from a crane.

The BSR provides high flux ($nv_{th} \cong 10^{13}/\text{cm}^2/\text{sec}$) at low cost and at reasonably low power (up to 1 megawatt). The cost for this reactor was low, primarily because of the absence of an external cooling system. Heat is removed entirely by natural convection. A force-cooling system could be provided, of course. Further elaboration of this trend would result in an MTR-type reactor (see Type III).

Access to the reactor is practically unrestricted. The use of water as both shield and coolant permits direct visual observation of the reactor core during operation; this is of considerable peda-

gogical advantage when the BSR is used by the Oak Ridge School of Reactor Technology.

The unlimited access to the side of the BSR allows the irradiation of large and bulky objects and the use of various large instruments against the reactor face. When beam tubes are used, essentially no restrictions to their size are introduced. Samples to be irradiated and detecting instruments must be enclosed in watertight housings, but this has not proved to be difficult. Even for housings that are not watertight, the water can be excluded by applying suitable air pressure to the housing.

The BSR is inherently safe for moderate excess reactivities (≤ 2 per cent $\Delta k/k$). Any violent power excursion produced by such moderate excess reactivities would be rapidly and safely stopped by expulsion of moderator from the reactor core. Gradual addition of larger excess reactivities would simply lead to boiling in the reactor core. Fission products in the fuel elements are contained within the fuel plates. No fission-product contamination of the pool water has been observed.

Building. The facility is housed in a steel-frame building with Q (corrugated metal) siding. The building is 77 by 51 ft over-all. It houses the pool and reactor in a bay 77 by 32 ft, and it also contains an office, two air-conditioned rooms, and a small shop. The general appearance and arrangement of the building is apparent from Figs. 2-2 and 2-3. An external storage building 20 by 40 ft is also available.

Pool and Water Supply. The pool is 40 ft long, 20 ft wide, and 20 ft deep; it is provided with a well 14 by 14 by 5 ft, centered 15 ft from the north end. One possible arrangement of the pool and reactor bridge is illustrated in Fig. 2-1. Figures 2-4 and 2-5 show cross sections of the pool.

An aluminum gate 12 by 21 ft can be lowered between guides located 10 ft from the south end of the pool, thus blocking off this end and permitting the rest of the pool to be pumped dry. Figure 2-6 is a view of this gate ready to be lowered into position.

Removable concrete blocks are provided as part of the pool floor. The blocks have a density

of 3.2 and exhibit good neutron- and gamma-attenuating properties. The reactor is frequently operated with all blocks removed.

As originally designed, the pool was filled by gravity flow from the laboratory process-water

The original water treatment consisted of adding 60 ppm of Na_2CrO_4 to the pool water. However, later operating experience indicated the desirability of additional treatment in order to minimize corrosion. Consequently a deminer-

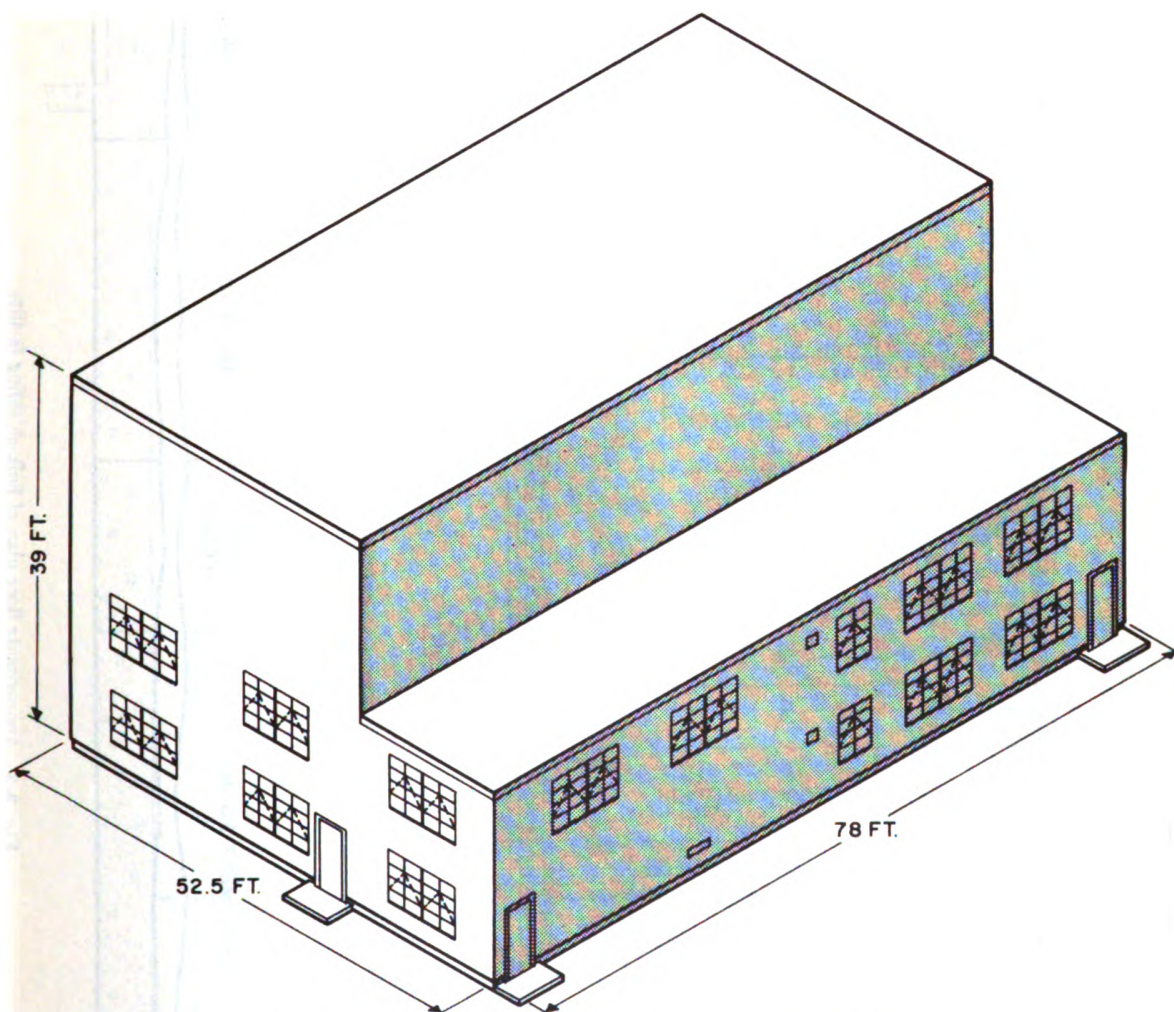
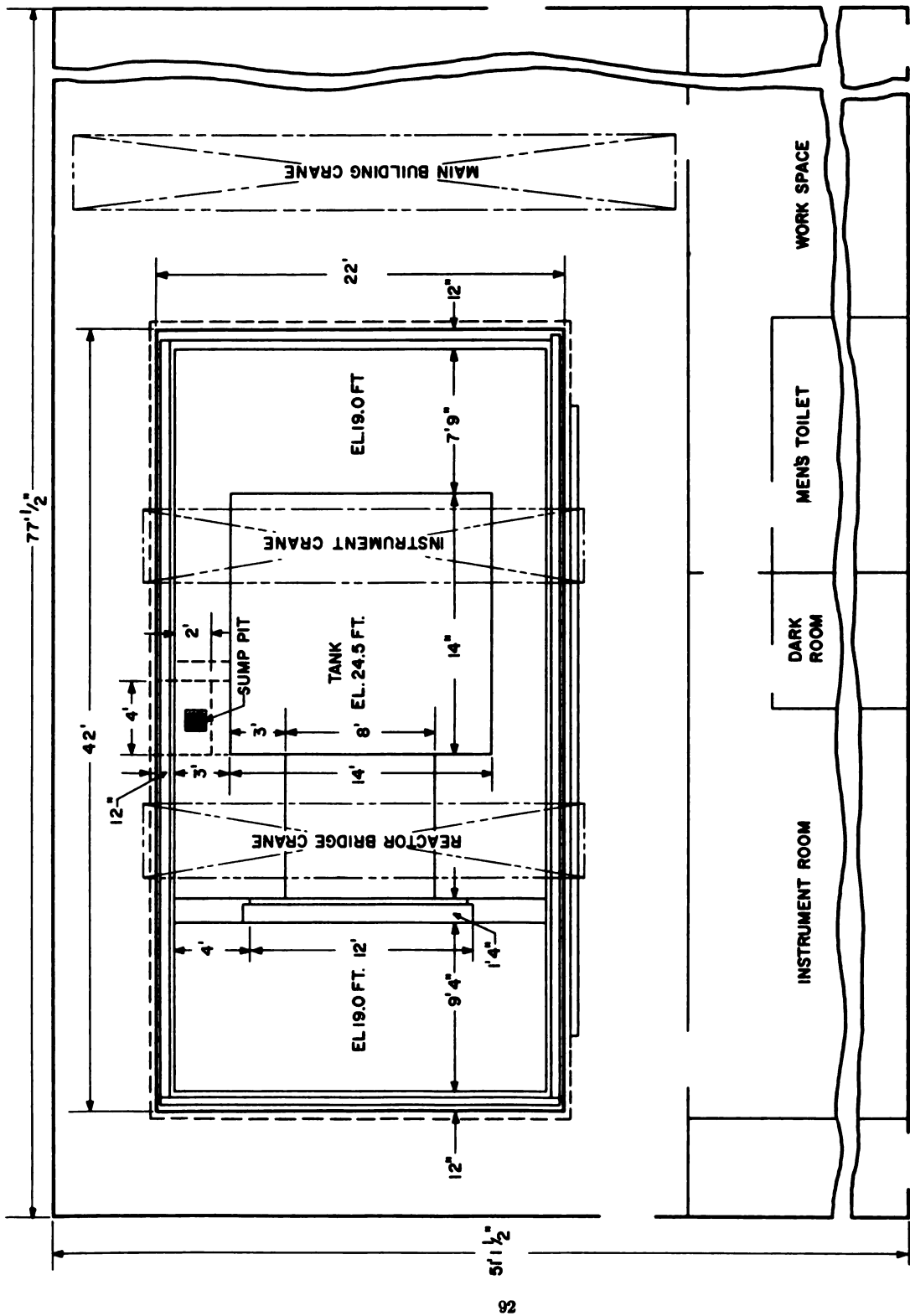


Fig. 2-2 Artist's sketch of building housing the bulk-shielding facility.

supply, and emptied by pumping. The outlet system has a vacuum breaker at its high point, which ensures that the pool will not empty by syphon action. The water drains into White Oak Creek and thence into the Clinch River. The pool has been emptied frequently for access, but not for safety reasons, and emptying into the creek and river has entailed no health hazards.

alizer was added to the system. This also improved visibility and radiation background. Pool walls are "Amercoated" to prevent removal of minerals from concrete. This white coating also improves visibility.

As originally designed, the reactor was suspended from a movable bridge by an aluminum frame so that the top of the active lattice was



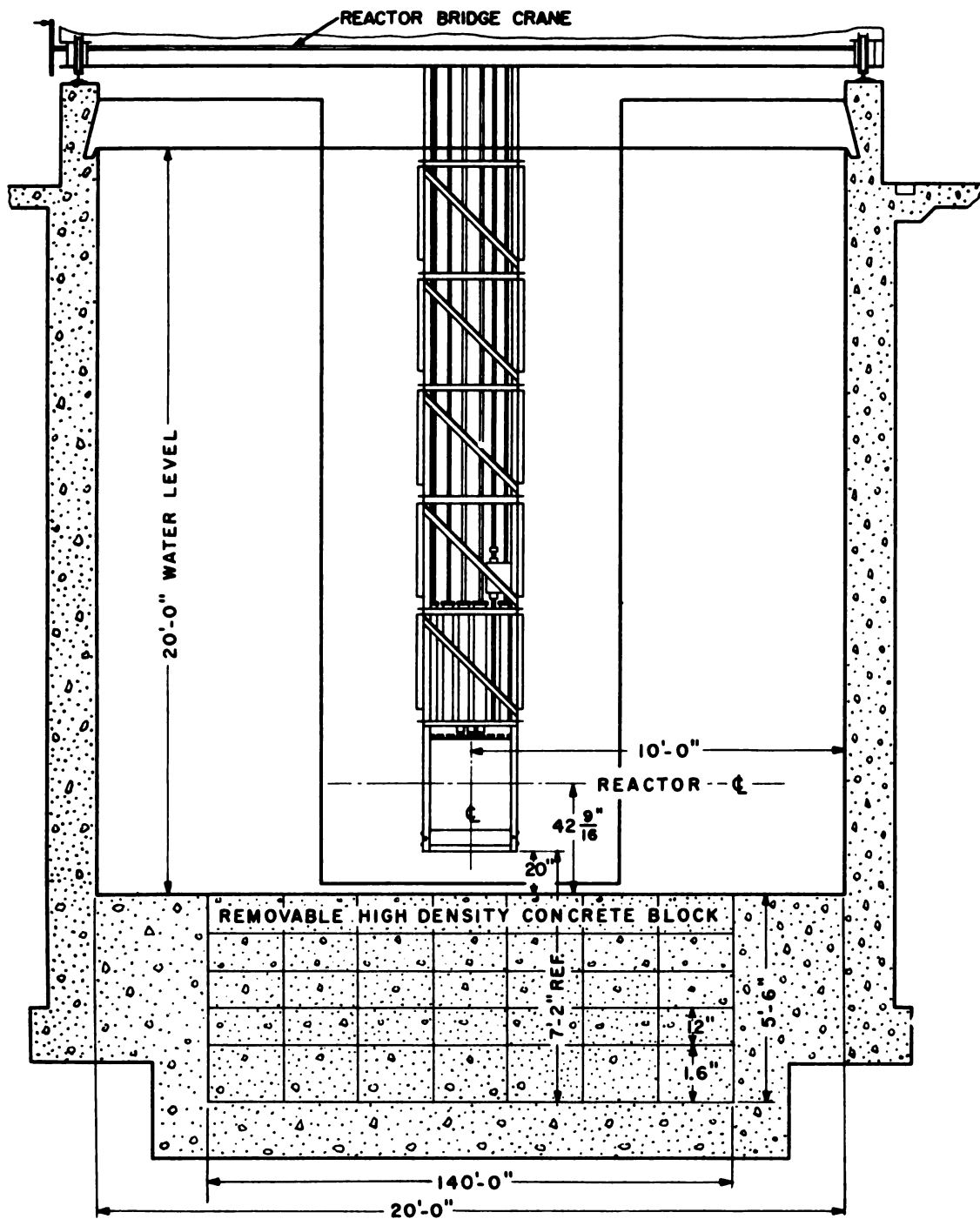


Fig. 2-4 Cross section of pool.

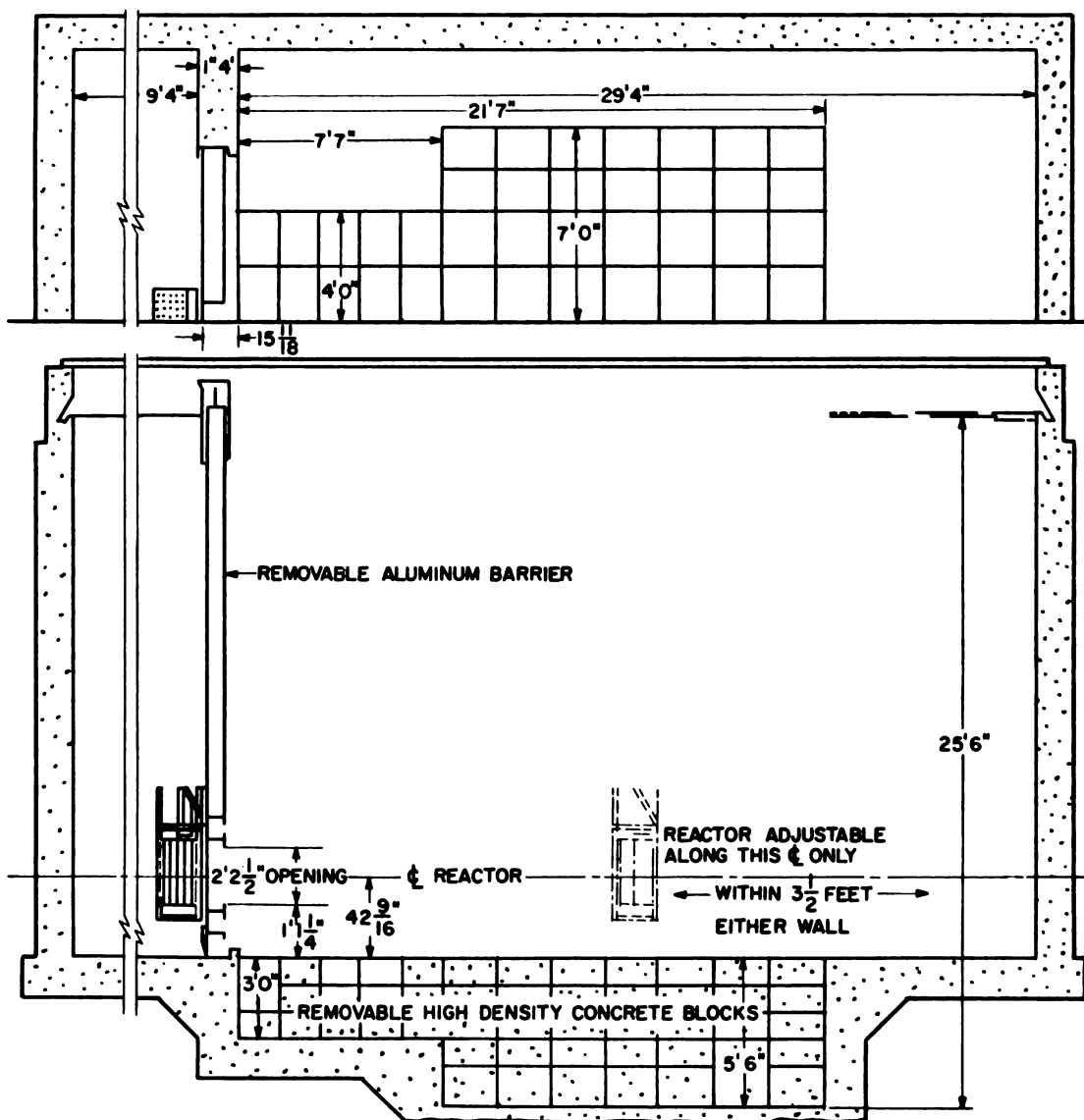


Fig. 2-5 Longitudinal section of pool.

15½ ft below the surface of the water. (A description of the reactor is given in the next section.) The reactor is currently operated with the water level at 1½ ft above the splash trough (shown in Figs. 2-4 and 2-5) in order to give a total of 17 ft of shielding for higher-power operation.

The heat capacity of the pool is sufficiently great to permit intermittent operation at more

than 100 kw without external cooling. The rise in temperature of the water with the reactor running at 100 kw is initially less than ½°F/hr. At 100 kw, the heat losses by conduction and evaporation are sufficiently high to stabilize the temperature at some reasonable level after the reactor has been operated for a brief period. If continuous operation at higher powers were desired, arrangements could be made to remove

heat either by adding cold (demineralized) water or by circulating the pool water through a heat exchanger.

The BSR has been operated as high as 1000 kw. However, at this power level, convection cooling water with its newly formed N^{16} comes to the surface too soon and constitutes a radiation

pool rather than on the bridge. When in operation, the reactor bridge is locked to the rails with heavy clamps. The reactor moves only along the long axis, north-south, center line of the pool.

The reactor itself is suspended by an aluminum framework from a pair of 12-in. I beams. The

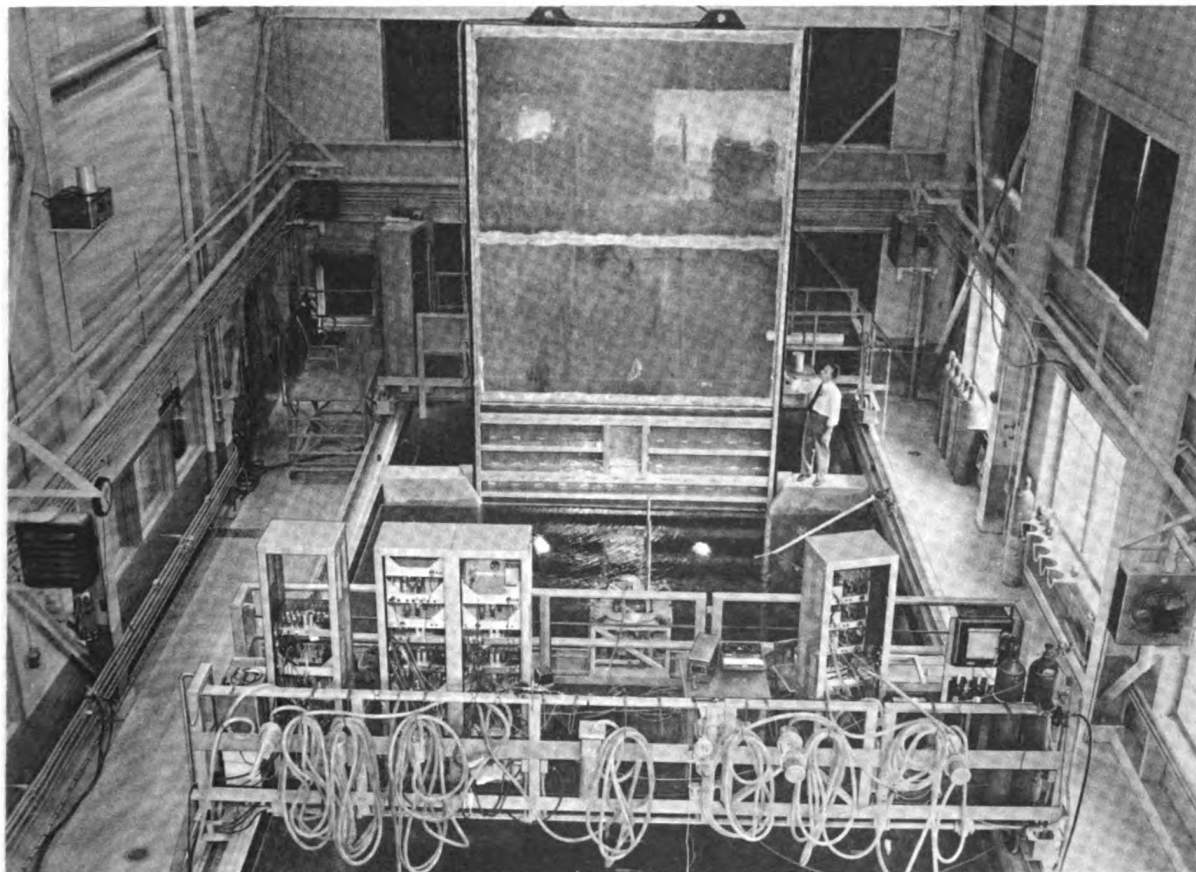


Fig. 2-6 View of pool and aluminum gate. The gate can be lowered to block off one end of pool, permitting the remainder to be pumped dry.

problem. This was eased to a great extent by using *jet diffusers* to divert and diffuse the heated, irradiated water emerging from the top of the core.

Reactor Supporting Structure. The reactor, controls, and control panels are all mounted on a bridge which spans the pool, and they move as a unit; the only connection to the rest of the building is through the power cables. The operator is located on a movable platform beside the

I beams have transverse bracing and are covered with a wooden platform. The supports for the framework are cantilevered out from one of the I beams, and the framework is completely open on one side. This enables an operator standing on the platform to remove or replace fuel elements in the reactor, with the aid of a specially designed long-handled tool, without draining the water from the tank. A $7\frac{1}{2}$ -ton overhead bridge crane is used in assembling the reactor, removing fuel elements, and similar operations.

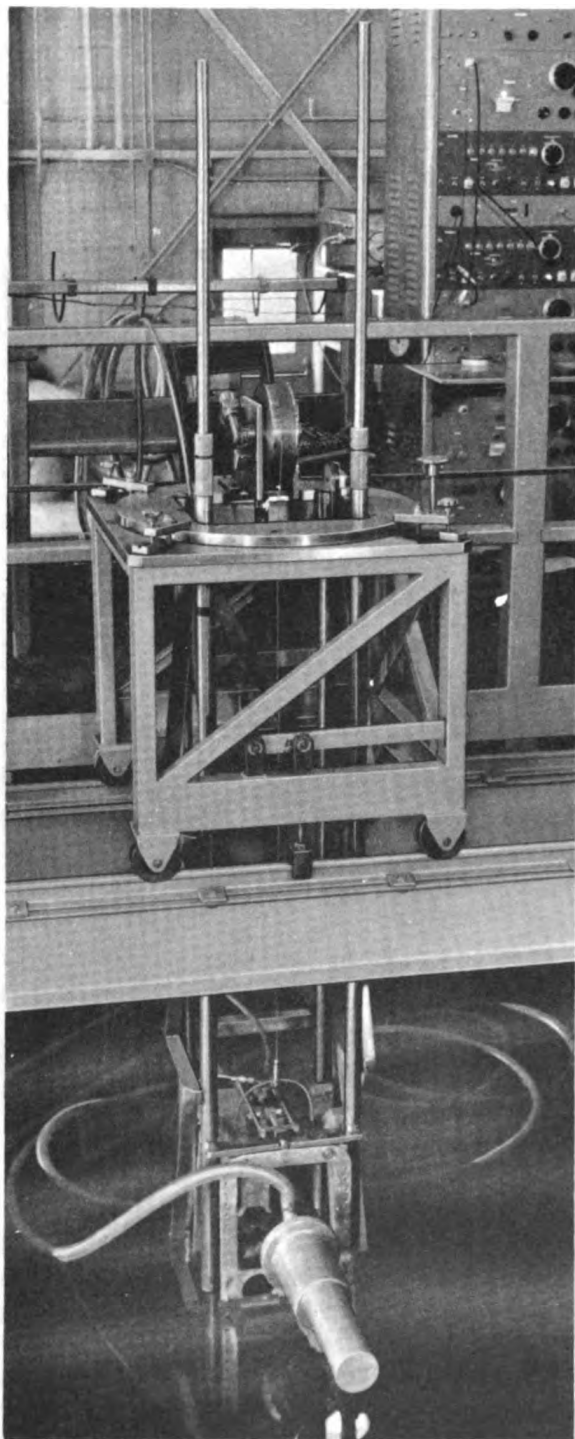


Fig. 2-7 View of instrument bridge and framework, on which measuring instruments may be lowered into the pool.

It is possible to roll the bridge and reactor to the small compartment at the south end of the pool, replace the aluminum barrier, and empty the rest of the pool. Personnel can then work on equipment in the empty section while being protected from radiation of fission products in the fuel elements.

Instruments and Instrument Bridge. An instrument bridge, somewhat similar to the reactor bridge, also spans the pool. One view of it is shown in Fig. 2-7. (It is also visible in the foreground of Fig. 2-6.) A cart travels on a pair of rails on one side of the bridge. A stainless-steel framework, suspended from the cart, reaches nearly to the bottom of the pool. A carriage which slides on the framework—and is raised and lowered with a winch—carries a number of diverse measuring instruments. The framework may also be rotated about a vertical axis. Thus the measuring instruments can be placed at any point in the pool and pointed in any direction in the horizontal plane. The bridge and cart can be clamped to the rails, and the carriage can be secured to the framework.

A list of the various measuring instruments on hand during the early days of the facility will be found in Appendix B. Some of these instruments are also shown in the drawings of Appendix C. The necessary power supplies, scalars, A-1 amplifiers, etc., were included as original equipment, and they are mounted on the instrument bridge. In most cases spares are provided to ensure continuity of operation.

No trouble has been experienced in operating the measuring instruments and associated preamplifiers under water. Each instrument is enclosed in a watertight aluminum or polystyrene case which is filled with compressed gas (CO_2 , argon, or dry air) at 8 psi above atmosphere. This will keep out the water if leaks should develop. The wiring leads are gathered into a cable and pulled through a length of Tygon tubing clamped to the case. An ionization chamber, with its preamplifier and cable, is shown in Fig. 2-8.

In order to measure distances under water, a sort of "traveling telescope" has been provided. This is a transit fitted with a long aluminum barrel and an optically flat glass window at the lower end, which is beneath the surface of the

water. Relative movements of the transit along the pool can be determined with the aid of a pointer and the scale which is fastened to the side of the pool. When the underwater lights are on, objects in the water are clearly visible.

Health-physics Instruments. A number of health-physics instruments are provided for personnel protection. These include three monitrons with boron-coated chambers which are sensitive to both gamma radiation and thermal neutrons. One monitron is mounted on the reactor bridge above the water and gives the dose rate to which operating personnel are exposed. Another monitron is mounted on the wall of the second-floor office area and affords protection to personnel there. (When the pool is drained, the reactor is not "visible" to people in the first-floor offices but can be "seen" from the second floor.) The third monitron is available for location in any suspected area. Additional health-physics instruments are listed in Appendix B.

REACTOR ASSEMBLY

The bulk-shielding reactor comprises an assembly of fuel elements, which is supported by a framework and grid and is filled with water. Furthermore, the reflector, coolant, and shield, which are separate components in most reactors, are here represented by the all-encompassing water of the pool. As can be inferred from the extensive use of water, the reactor is of the thermal type.

Fuel Elements. The fuel elements are made up of three main components. Fuel plates and side plates are welded together to form a rectangular pipe with a conical end section welded to the bottom. A photograph of the fuel elements is shown in Fig. 2-9, and drawings appear in Appendix C.

The standard fuel element is about 3 by 3 in. in cross section and 35 in. long including the end section. The active section is only 24 in. long, however. In the BSR design the square-lower-end boxes of the original MTR elements are replaced with round-end boxes, and the upper-end boxes are omitted.

The standard fuel element contains 18 convex

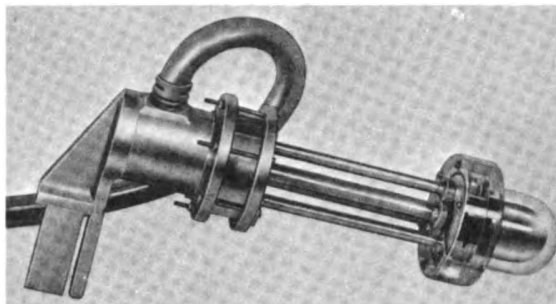


Fig. 2-8 Ionization chamber and preamplifier for underwater use.

fuel plates, each 3 in. wide, 24 in. long, and 0.060 in. thick. The uranium in the fuel plate is in a highly enriched form, with each standard fuel element containing about 140 g of U^{235} . Taking into account the structural members, the aluminum-to-water ratio is about 0.7. Each fuel plate consists of a sheet of aluminum-uranium alloy sandwiched between two 2S aluminum sheets. This assembly is hot-rolled into a solid plate of the proper dimensions. It is sufficiently tight so that fission products cannot escape. The plates are clad with a thin layer of 72S aluminum, which corrodes preferentially.

The control-rod-receiving fuel elements are similar to the standard element except for slots provided for the entry of the control rods. These slots are obtained by removing half of the fuel plates, thus providing apertures of $1\frac{1}{8}$ by $2\frac{5}{8}$ in. The rods have cross sections of about $\frac{1}{8}$ by $2\frac{1}{4}$ in. and readily move inside the fuel elements.

Fuel (U^{235}) is consumed at a rate of about 1 g per 24 megawatt-hr (1 megawatt-day) of operation. The fuel elements must be reprocessed when 5 to 10 per cent of the fuel (approximately 140 g) has been consumed. Thus, at the normal power level the maximum life is a little less than 1 megawatt-year. At power levels no greater than 100 kw, the effect of poisoning due to fission products can be neglected. Unless care is taken, however, the life of the fuel elements may be determined by corrosion rather than by poisoning.

When the reactor is operated at high power, the long-life fission products accumulate, and the fuel elements will not "cool" (radiationwise) in a reasonable time. The elements can then be removed from the pool only inside a protective shield. After long periods of operation at 100 kw,

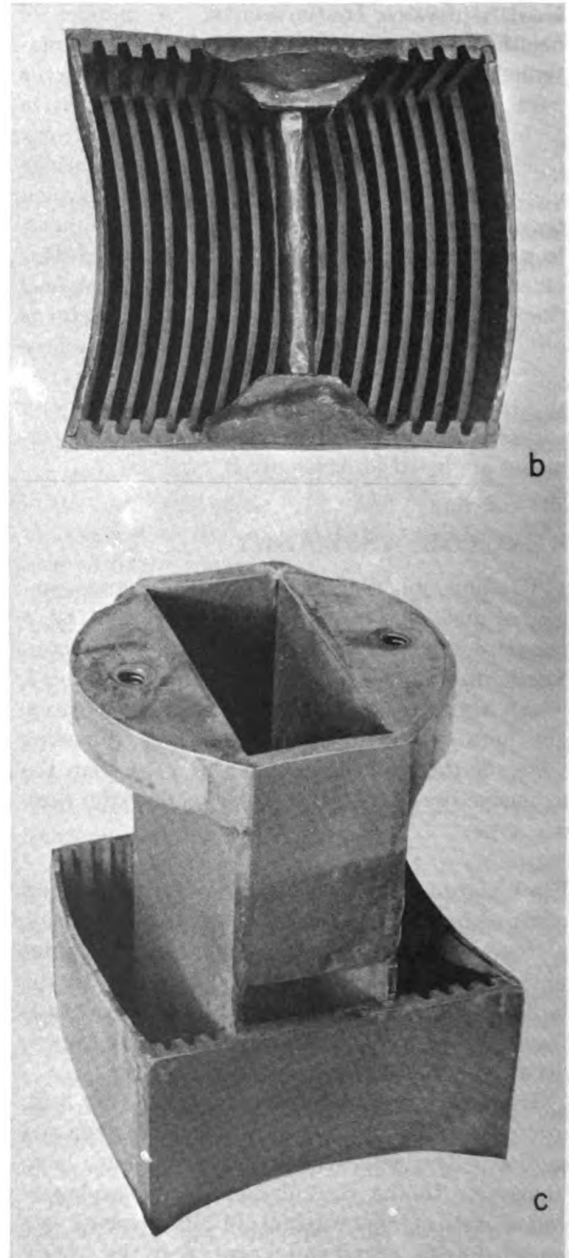
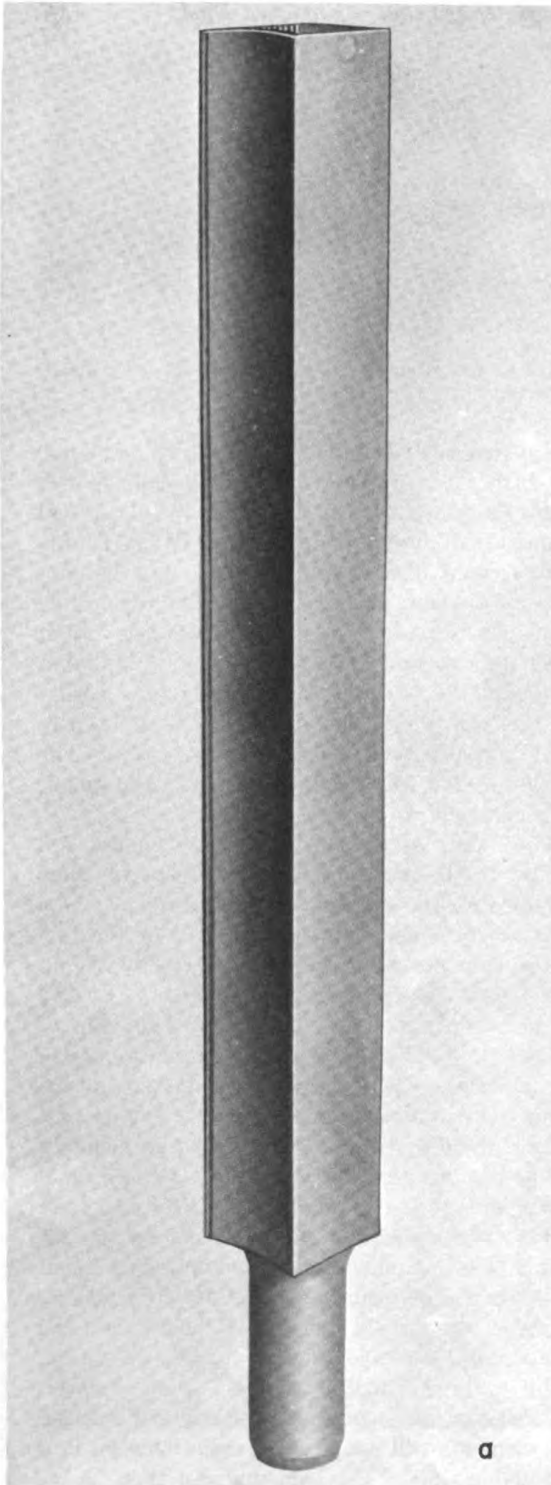


Fig. 2-9 Fuel elements. *a*, side view of standard fuel element; *b*, top view of standard fuel element; *c*, top view of control-rod fuel element.

the reactor must be shut down for a few days before the elements may be removed. A lead "coffin" with walls 4 in. thick will then give adequate protection from one element. Such a shield weighs a little less than a ton. A second set of fuel elements, for use at power levels at about a watt or less, has also been provided. These are used for low-level measurements, where high gamma activity from fission products is to be avoided.

Grid Plate and Core Stacking. Fuel and other elements rest in an aluminum grid with dimensions of 28 by 19 by 5 in. A total of 54 holes permits loading of any array up to 9 elements by 6 elements. Actually, criticality is reached with considerably fewer elements, but this grid design permits flexibility in the loading pattern. Figure 2-10 is a photograph of the grid structure; the

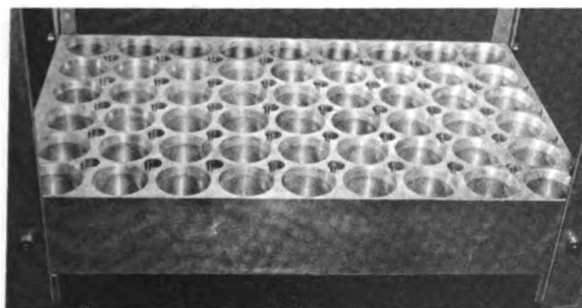


Fig. 2-10 Aluminum grid plate; the pins between each fuel element prevent the element from rotating.

pins visible between each element prevent the fuel element from rotating. Figure 2-11 is another view, showing the grid and support structure before loading. The cylindrical aluminum cans above the grid contain the ion chambers. In front of the cans are the electromagnets which support the safety and control rods. Figure 2-12 shows the same reactor with the three special control-rod-type fuel elements in place. Figure 2-13 shows the reactor completely loaded. The elements visible in front are the BeO-filled reflector units.

With the aid of a special tool, fuel and reflector units can be loaded or unloaded from the grid by an operator standing on the reactor bridge. This tool has a handle about 20 ft long, with a hook at the lower end and an operating lever at the

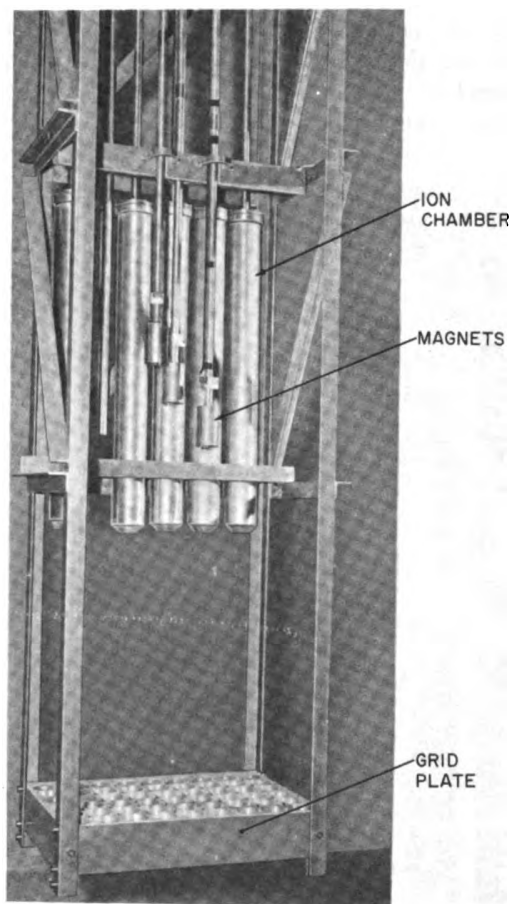


Fig. 2-11 Reactor grid and support structure before loading.

upper end which releases the hook. The fuel elements can be handled under water at all times, thus protecting the operator from radiation by fission products. When not in use, the "hot" fuel elements are stored in a rack under 8 ft of water.

Reflector. The reflector normally consists of the pool water adjacent to the core. However, other simple arrangements are possible. If desired, dummy fuel elements can be filled with graphite, beryllium oxide, or other material as desired, and inserted in the grid around the core proper. Or, as was done during initial operation of the reactor, a large slab of reflector material can be placed along one or more sides of the core assembly.

A good reflector placed around the active lattice will reduce the fuel requirement and improve the flux distribution in the core. A satis-

factory scheme used on the BSR is to provide a number of aluminum cans of the same outline as a fuel element and filled with cold-pressed beryllium oxide bricks (see Fig. 2-13). A reflector composed of a 3-in. thickness of BeO on the four

is essentially infinite. Shielding in a vertical direction is accomplished by the 17 ft of pool water between the core and the surface, as has been noted before. Even with only $15\frac{1}{2}$ ft of water between the core and the surface, gamma

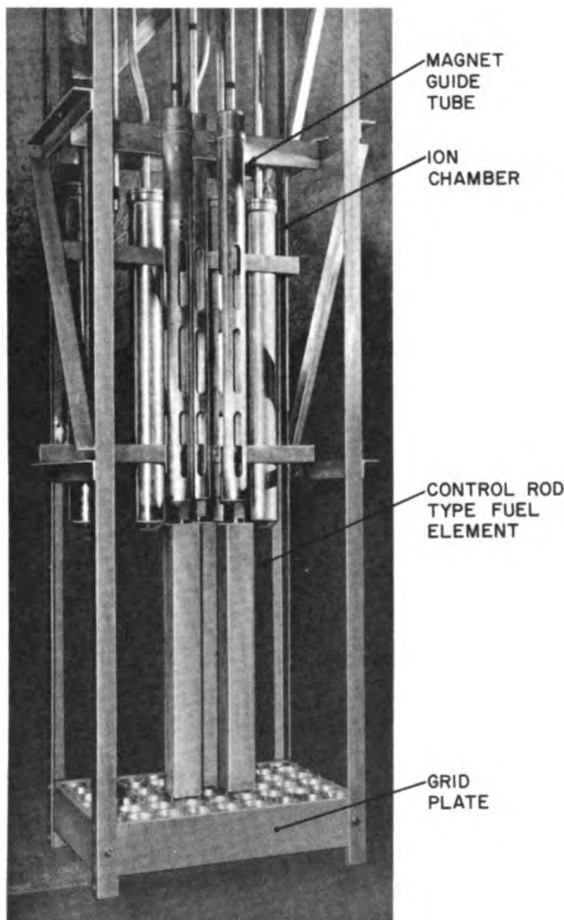


Fig. 2-12 Reactor partially loaded, with the three special fuel elements in place.

vertical sides will reduce the size of the active lattice to an array of four elements by five elements and the critical mass to about 2.5 kg. Thirty cans of BeO are sufficient to surround the BSR lattice. The BeO reflector blocks used at ORNL were cold-pressed in the Bureau of Mines Laboratory in Norris, Tennessee. These blocks have an average density of 2.3 compared to the theoretical value of 3.0.

Shield. The reactor core is located below ground level so that the shielding in a horizontal direction

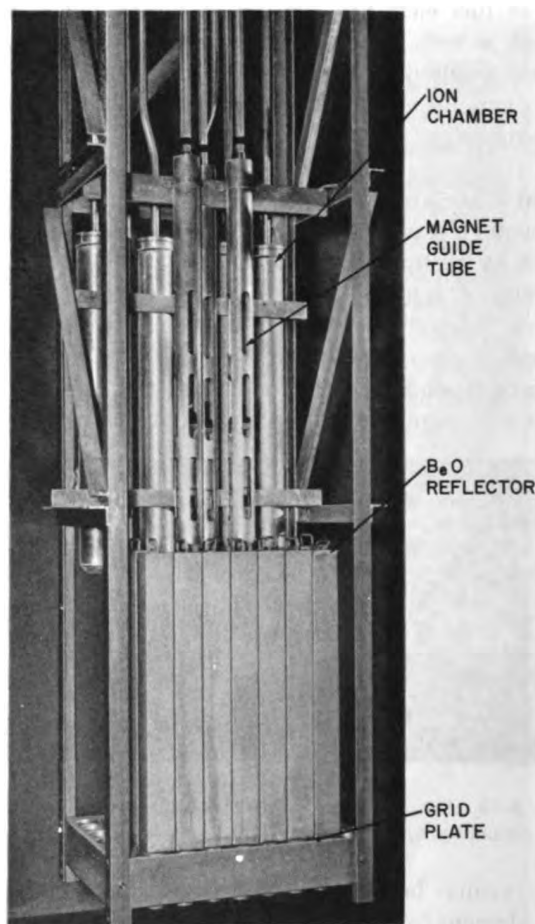


Fig. 2-13 Reactor completely loaded. The elements visible in front are the BeO-filled reflector units.

rays are sufficiently attenuated so that a person standing next to the pool will receive less than 60 mr in 8 hr when the reactor is operating at 100 kw. The amount of water between the core and the walls of the pool is also adequate to prevent the concrete from becoming appreciably activated by neutron absorption.

CONTROL AND SAFETY SYSTEM

Control Rods and Actuators. The reactor is provided with two boron-lead shim-safety rods

consisting of a mixture of lead and boron carbide enclosed in an oval aluminum can $\frac{7}{8}$ by $2\frac{1}{4}$ by 26 in. long (see Fig. 2-14). These rods travel in special fuel elements, which have longitudinal holes $1\frac{1}{8}$ by $2\frac{5}{8}$ in. (see Fig. 2-9). Having nearly the density of lead, the rods fall with an acceleration close to gravity; the fall is damped by spring-type shock absorbers. An iron armature is fastened to the top of each rod, and this in turn is suspended from an electromagnet which can be raised or lowered with the aid of a small electric motor. A drawing showing these components will be found in Appendix C. Each rod is equivalent to between 2.5 and 3.8 per cent $\Delta k/k$, depending on the loading. A control or regulating rod of the same dimensions and construction as the shim-safety rods, but containing only 8 g of boron (equivalent to about 0.6 per cent $\Delta k/k$), was in use; it has been replaced by a rod which depends only upon stainless steel for neutron absorption.

Each electromagnet which supports a safety or a control rod is an ironclad type made from one piece of Armco soft iron. The exciting coil has 4800 turns of No. 30 copper wire and is impregnated under vacuum with special varnish (Irvington) and oil impregnation (Insulator Company's Harvel "Oil Stop"). This forms a waterproof and shock-absorbing bulk insulator around the coil. Tests in the ORNL graphite pile indicate that this material is stable under neutron and gamma-ray irradiation. As an added precaution against moisture all joints are painted with Glyptal.

The magnets were designed to support the rods with a 0.05-in. air gap and with a 30-ma exciting current in the windings. The air gap is obtained by crowning the face of the armature. The crowning also makes perfect alignment between magnet and armature faces unnecessary and obviates the need for a universal joint. Tests indicate that the release time, with 50 per cent more current than is required to hold the control rod, is of the order of 100 msec. Since the exciting current is an inverse function of the reactor power level, the actual release time after the reactor passes the scram level is much less than this.

The magnets (and hence the safety and control rods) are raised and lowered by steel wires which

travel up to motor-driven drums on the reactor bridge. Although the lack of positive drive in the downward direction has some possible advantages (the rods cannot punch holes into misaligned fuel elements), this type of rod drive is not being favored in newer models.

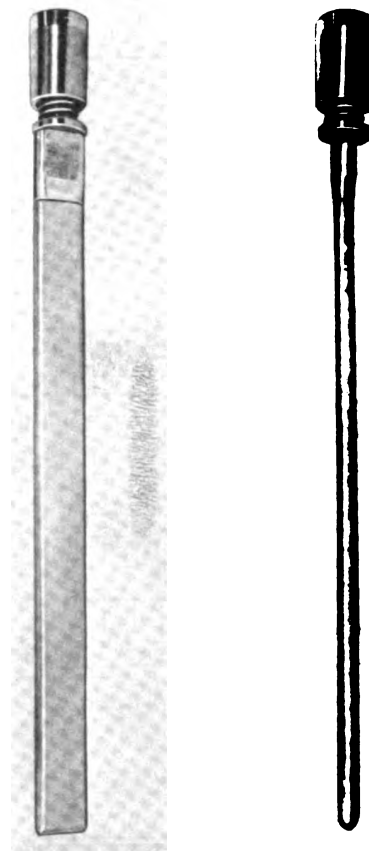


Fig. 2-14 Boron-lead shim-safety rod; view at right shows rod rotated through 90° along its axis.

Control and Safety Instruments. Sensory instruments for control and safety include differential chambers, boron-coated safety chambers, a fission chamber, and a gamma chamber. A monitor with a boron-coated chamber is also connected into the safety circuit.

The differential chamber is a 4-in. model which contains two chambers, one of which is boron-coated. Both chambers respond to gamma rays, but only the boron-coated chamber responds to neutrons. The outputs are connected in opposition so that, with proper compensation, the net

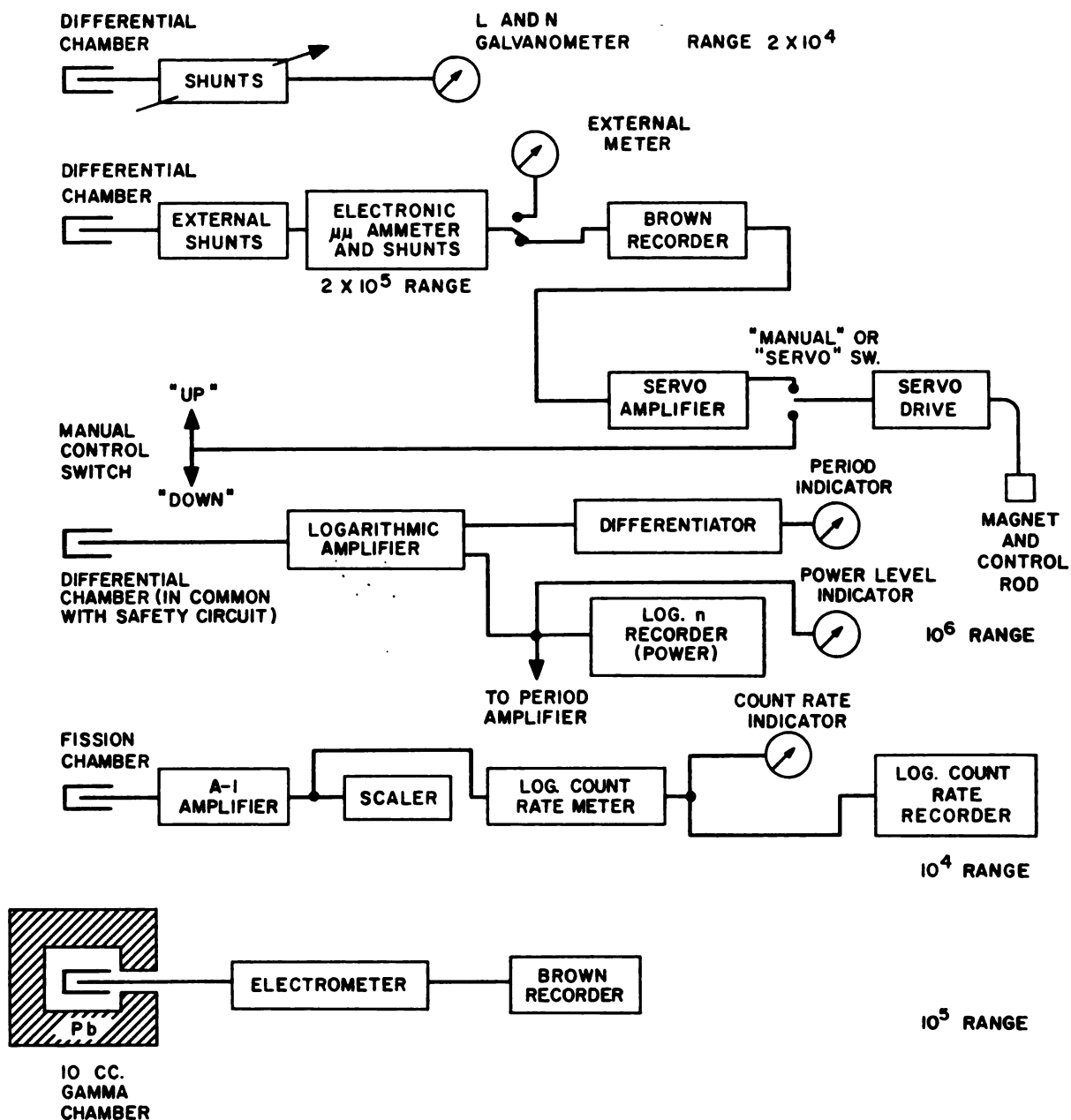


Fig. 2-15 Block diagram of control circuits.

output is a function of the neutron flux only. The chamber is illustrated in Appendix C.

To provide indication during startup, a U^{235} fission chamber, an amplifier, a scaler and register, and a log count-rate meter and recorder are provided. With the rods down, the source is of sufficient strength to supply about 3 counts/sec. This circuit has a range of 10^4 (10,000 counts/

sec maximum). Both the micromicroammeter and the log N circuits respond before this level is reached. An electric-motor drive raises the fission chamber to keep it from being activated at high reactor powers.

A 10-cm³ graphite gamma chamber is placed against the rear face of the reactor. A lead half cylinder 3 in. thick shields the chamber from cap-

ture gammas in the water. Thus the chamber current is very nearly proportional to the pile gammas. The chamber is connected to an electrometer and a Brown recorder. Some arrangement such as this is required to give an accurate measure of the gammas from the reactor, since the gammas from fission products will be appreciable compared to gammas resulting from fissions when pile power is reduced after operation at high levels. This instrument is also illustrated in Appendix C.

The safety chambers are 3-in. boron-coated ionization chambers which supply a current proportional to the neutron-plus-gamma level. This current flows through a high resistance across the input of a preamplifier, consisting of a single-stage cathode follower.

Control Circuits. A block diagram of the circuits for controlling and observing the operation of the reactor is shown in Fig. 2-15. The first instrument is one of the 4-in. differential chambers which supplies a Leeds and Northrup model 2430D galvanometer. There are no electron tubes in this circuit, and the galvanometer reading is always directly proportional to the chamber current. The range is about 2×10^4 , and the relation between the galvanometer indication and the reactor flux depends on the location of the chamber.

Over the lower portion of the operating range the BSR is essentially a constant-temperature reactor. Hence the temperature does not vary sufficiently to stabilize the reactor, and a servo control is supplied to hold the power level constant. This servo system is more than adequate for its present application.

A differential chamber is connected to a Leeds and Northrup model 983A electronic micromicroammeter. With the aid of shunts, the range is extended to 2×10^5 . The micromicroammeter controls a Brown recorder to which an extra slide-wire has been added. The electrical difference between this slide-wire slider and a demand reference controls the servo.

A third differential chamber supplies a logarithmic amplifier. The output of this amplifier is differentiated to give indication of reactor period. The output also controls a Brown recorder, from which a continuous record of the power level

over a range of 10^6 is available. This is the so-called *log N indicator*.

The electronic equipment for operating and monitoring the reactor is contained in five relay racks, of which three form the control panel (Fig. 2-16). The other two contain power supplies, amplifiers, and relays for the control interlocks. The operating panel (the section with a sloping front)

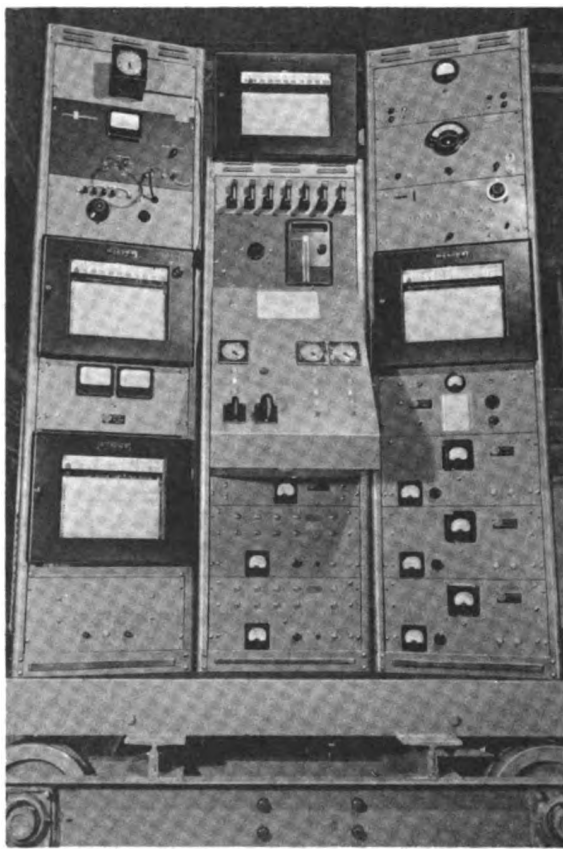


Fig. 2-16 View of reactor control panel.

has two switches for raising and lowering the safety rods, selsyn indicators to show the rod locations, a switch to raise and lower the fission chamber, a MANUAL or SERVO switch to activate the automatic control system, and a switch to operate the control rod when the circuit is in MANUAL position. A SCRAM button and a key switch are also provided. The SCRAM button, in addition to dropping the rods, resets the safety circuit after it has been tripped. Flowmeters (appearing near the top of the center panel) are placed in the gas lines of all chambers requiring

gas flow, and enable the operator to determine whether or not the gas flow through the chambers is satisfactory.

Safety Circuits. In addition to the control system, there is an essentially separate safety system. The philosophy of the safety system is that it should "fail safe"; i.e., the safety rods must

a common point known as *sigma bus*. The differential chamber is identical with the one shown in Fig. 2-15, which feeds into the period indicator. As part of the safety circuits the chamber also feeds through a period amplifier to a sigma amplifier and thence into the sigma bus.

The electromagnets which support the safety and control rods draw exciting current from sepa-

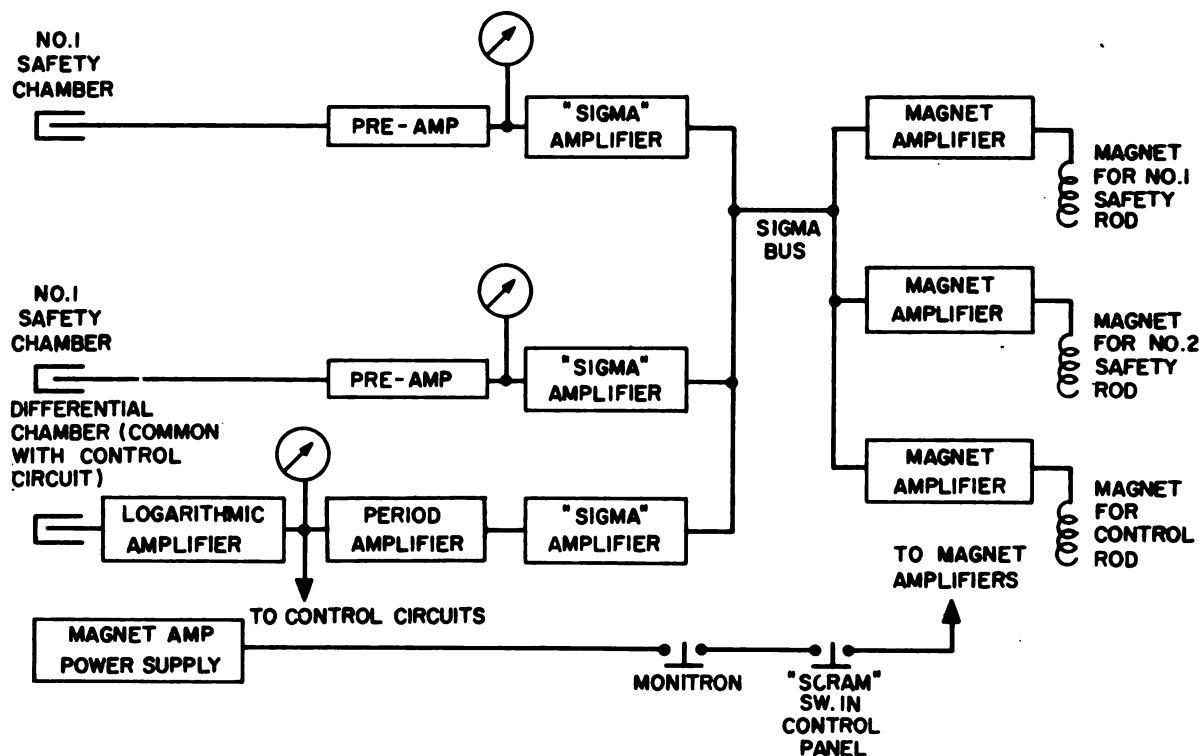


Fig. 2-17 Block diagram of safety circuits.

fall if the power is cut off or if major circuit trouble develops. In furtherance of this idea, vacuum tubes are used throughout in preference to gas tubes or relays. The amplifiers are monitored at several points and lights warn the operator of an abnormal condition. The safety system is a modified copy of the one which has been extensively tested in the low-intensity test reactor (LITR) at Oak Ridge. A block diagram of the system is shown in Fig. 2-17.

As is apparent from the block diagram, the output of each safety-chamber preamplifier feeds a d-c amplifier (the *sigma* amplifier) which is the source of the signal for operating the safety circuits. Both of these amplifiers are connected to

rate magnet amplifiers. The inputs to these amplifiers are controlled by the voltage on the sigma bus; as this voltage is increased, the current through the magnets decreases. Thus, the result of an increase in neutron flux is to reduce the magnet current. The circuit is adjusted to release the rods when the flux reaches a predetermined level. (Grounding the sigma bus will also cause the rods to drop.)

Adjustment is obtained by changing the pre-amplifier input resistance or moving the safety chambers away from the reactor. The speed of operation of the electronic portion of the circuit is determined by the time constant of the chamber, cable, and input resistor. Actually, most of

the delay in operation is associated with the inductance of the electromagnets and the finite time required to accelerate the dropping rods.

In this facility the reactor operator must keep closely informed concerning the progress of the measurements. Because of this, the circuits are not as thoroughly interlocked as those in the MTR, for example. Nevertheless a certain amount of interlock-type protection is provided. The safety rods are raised by three-phase a-c motors with a definite speed, thus ensuring against too rapid withdrawal. It requires about 2 min to raise the rods completely. The reactor is operated only when the safety rods are withdrawn at least three-fourths of the way. This ensures that a negative $\Delta k/k$ of 5 to 7 per cent is always obtainable by dropping the rods.

An interlock, actuated by the pointer of the recorder in the fission count-rate circuit, keeps the operator from raising the safety rods unless the count rate is more than 2 per sec. This prevents startup unless the polonium-beryllium source and the fission chamber are in place. In addition, the magnet current will be interrupted if the monitor fixed on the bridge shows more than a few milliroentgens per hour of gamma rays emerging from the water, or if one of the SCRAM buttons located on the control panel and on the walls near the four corners of the pool is operated. When the magnet-current circuit is completed, REACTOR ON signs located near the pool and above all the entrances are lighted automatically.

PERFORMANCE AND OPERATING CHARACTERISTICS

This section includes experimentally determined data on the performance of pool-type reactors, which may be helpful in the design of new models. Data on the radiation characteristics of the BSR include quantity as well as energy spectra of the leakage radiation. Considerable information on water-shield performance has been made available since the BSR was originally designed for testing shields. Information is also included on some types of critical loadings.

Typical Critical Loadings. A great many different configurations have been loaded into the BSR. Although a number of these loadings util-

ized BeO reflectors, the present emphasis is on water-reflected reactors. Figure 2-18 shows several typical loadings.

Control-rod Effectiveness. Experimental data on the effectiveness of control rods have been obtained for several control rods and in several configurations. The results are tabulated in Table 2-1. Control and calibration methods are described in the section Experimental Methods.

Table 2-1 Effectiveness of Control Rods

Control rod	% $\Delta k/k$ (for whole rod, except as noted)		
	Loading 5 × 6 H ₂ O- reflected	Loading BeO- reflected (Fig. 2-18b)	Loading H ₂ O- reflected (Fig. 2-18a)
Cd-Pb regulating rod *	0.6		
Cd-Pb safety rod †.....	1.3		
Stainless-steel regulating rod ‡.....	...	1.2	0.73
B ₄ C safety rod ¶.....	...	3.8	2.5

* An aluminum shell filled with a molten mixture containing 7½ g cadmium and the remainder lead. The effectiveness was measured over the last 12 in. of travel.

† An aluminum shell filled with a molten mixture containing 17½ per cent by weight of cadmium.

‡ A cylinder of type 347 stainless steel with 0.087-in.-thick walls.

¶ An aluminum shell packed with B₄C powder.

Temperature Coefficient. Temperature coefficients have been obtained by heating the water in which the reactor was suspended and measuring the effect on reactivity by means of a calibrated control rod. The temperature coefficient, indicated by the change in reactivity in Fig. 2-19, is not constant, even over the small temperature range investigated; at 26°C the temperature coefficient is -6×10^{-5} , while at 55°C it is at least -10×10^{-5} . If only the end points of the curve are used, the average temperature coefficient is -8×10^{-5} per °C. For additional details see the section Experimental Methods.

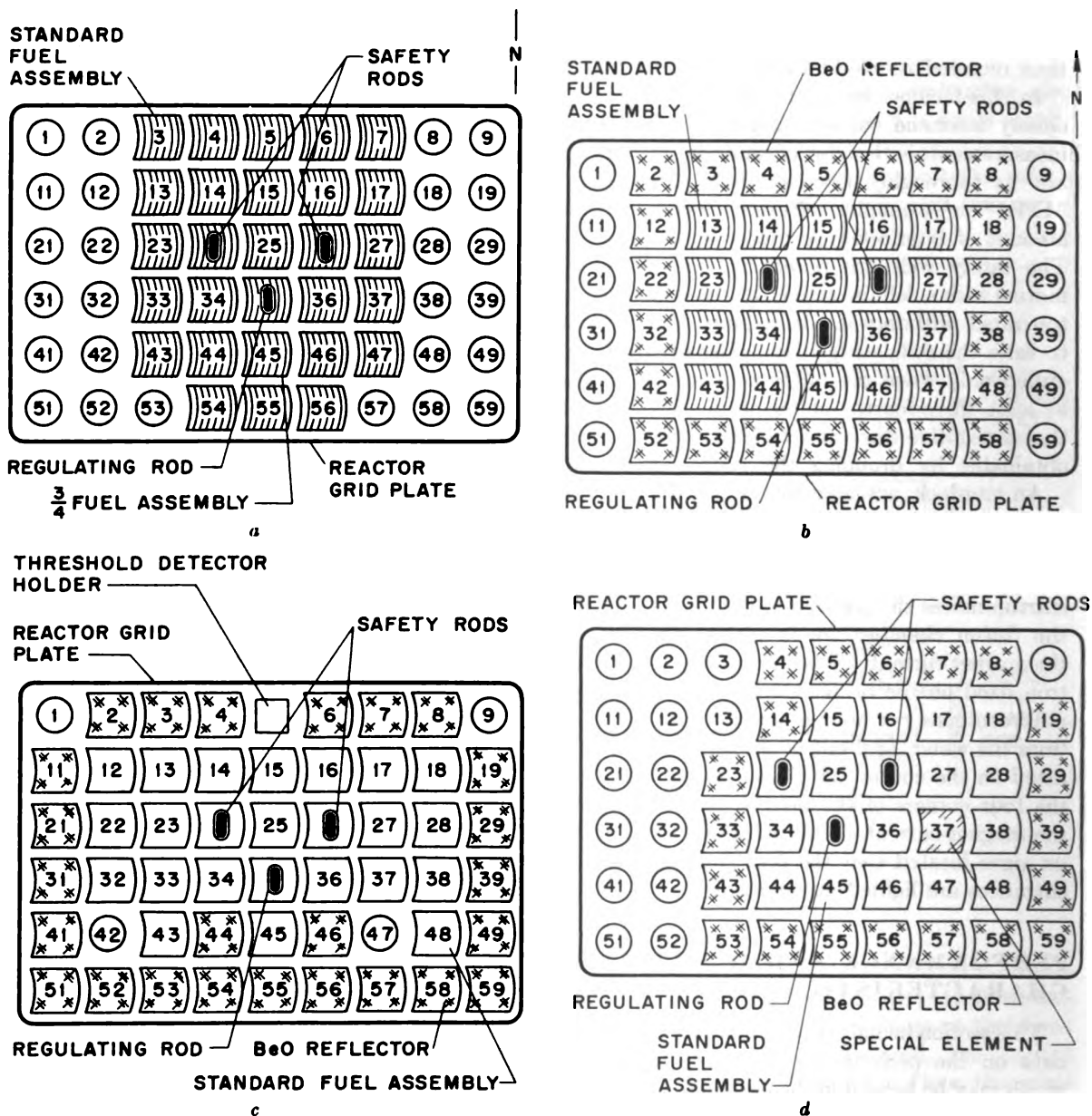


Fig. 2-18 Typical critical loadings. *a*, water-reflected, U^{235} load 3.7 kg; critical load 3.5 kg; *b*, BeO reflected, U^{235} load 2.6 kg; *c*, configuration for radioactivation, U^{235} load 3.1 kg; *d*, configuration for foil exposures, U^{235} load 2.4 kg.

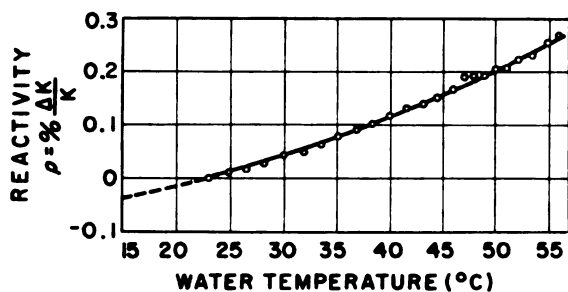


Fig. 2-19 Temperature coefficient of the BSR.

Neutron Spectrum of Reactor. The neutron spectrum in a fuel element of the BSR was measured by means of radioactivants. The reactor fuel loading shown in Fig. 2-18c was used for this series of measurements. An attempt was made to minimize local neutron-flux fluctuations at the site of measurement by holding the nearest con-

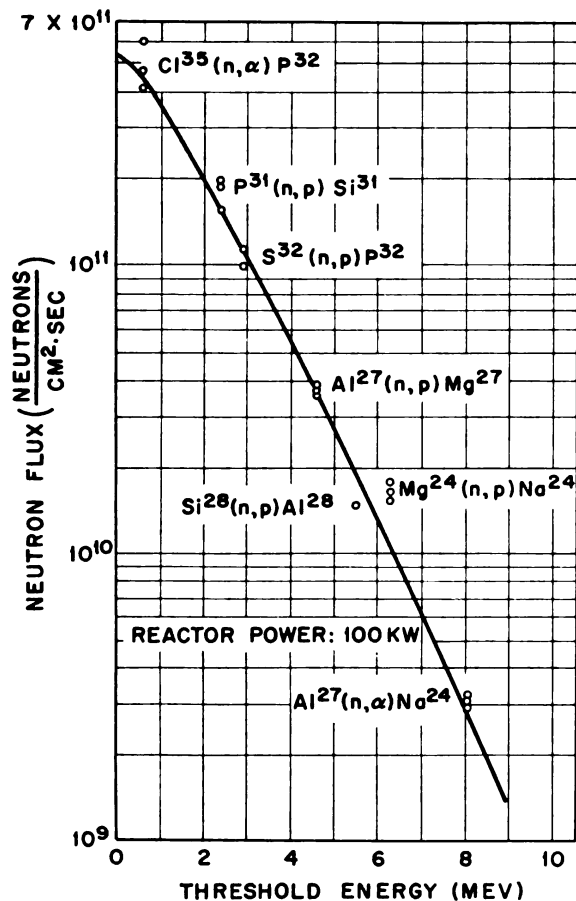


Fig. 2-20 Neutron flux above threshold as a function of threshold energy.

trol rod (more than 6 in. distant from the measurement) in an approximately constant position. All measurements were made for a reactor operating power of 100 kw.

The results are shown in Figs. 2-20 and 2-21, which give spectral data for the energy regions above 0.5 Mev and below 0.1 Mev, respectively. Figure 2-20 is a graph of the neutron flux above threshold energies as a function of threshold energies. The curve is taken from Watt's measurements of the fission spectrum and is normalized

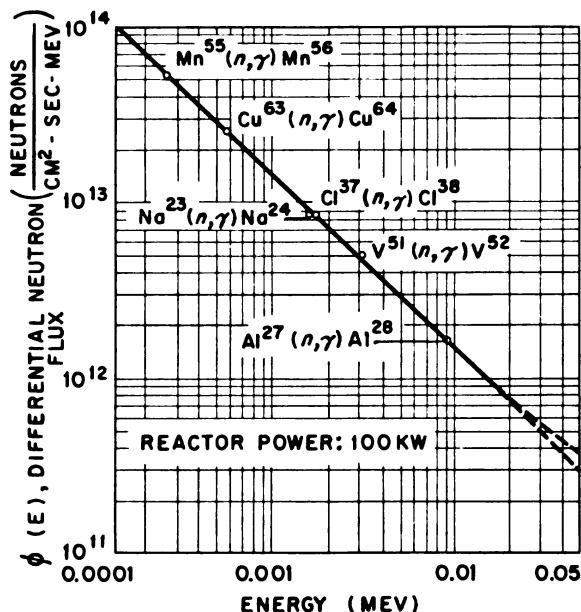


Fig. 2-21 Neutron energy spectrum for the epithermal energy region.

to the present data at the threshold energy for the $\text{Al}^{27}(n,p)\text{Mg}^{27}$ reaction. Data for the low-energy region (Fig. 2-21) are in good agreement with a $1/E$ spectral function.

Flux Traverses. *Thermal-neutron flux.* Flux measurements have been made throughout the BSR core by exposing gold and cobalt foils with and without cadmium covers, the difference in activation being proportional to the thermal-neutron flux. Absolute flux measurements with gold detectors are based on the calibration of the ORNL standard graphite pile. Absolute flux measurements with cobalt detectors are determined from the absolute disintegration rate of the

sample, as measured in a four-pi-gamma ionization chamber.

Figure 2-22 shows vertical traverses on two opposite faces and down the the middle of a BSR fuel element. Peaking in the reflector is apparent at the ends of the fuel elements (12 in. from center line). Figure 2-23 illustrates north-south ther-

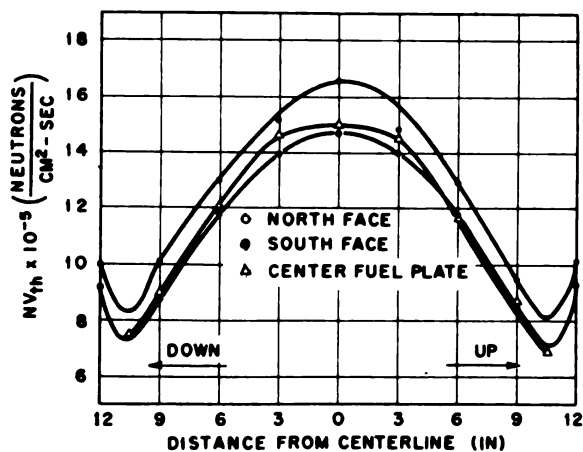


Fig. 2-22 Vertical flux traverses.

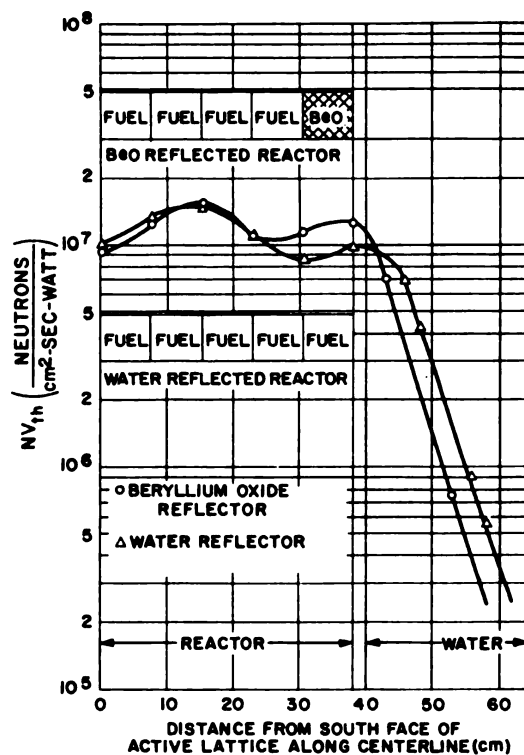


Fig. 2-23 Horizontal (north-south) flux traverses.

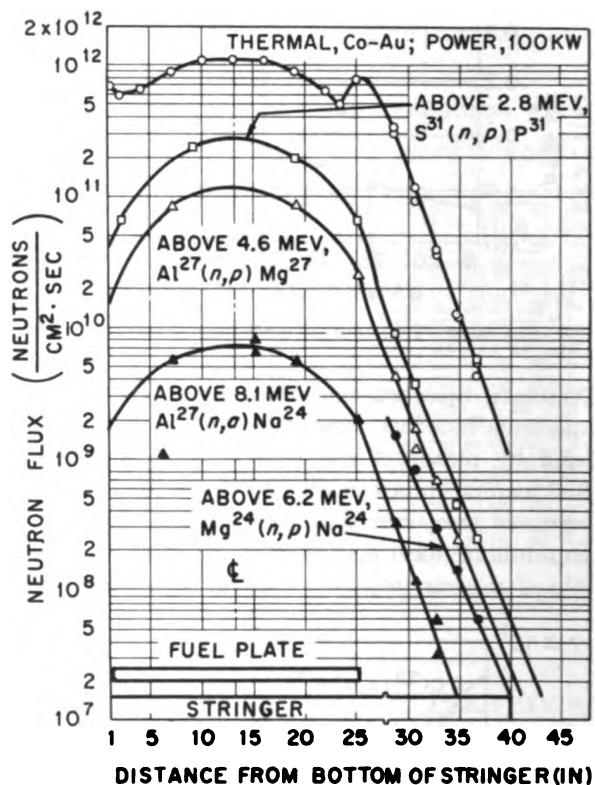


Fig. 2-24 Flux traverses through BSR fuel element.

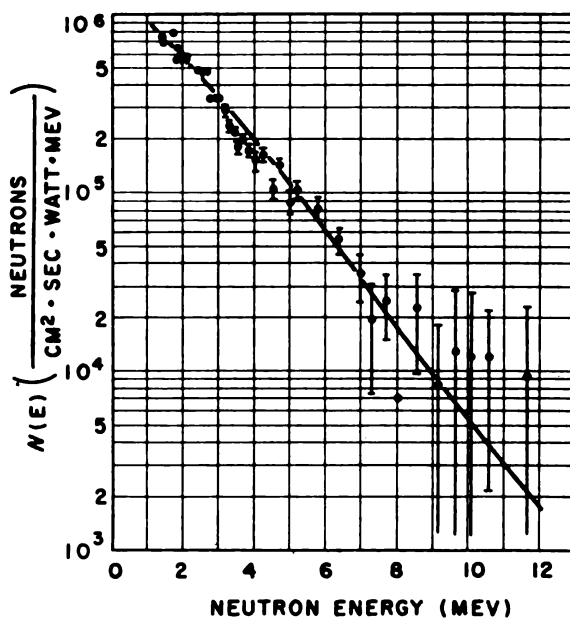


Fig. 2-25 Fast-neutron leakage spectrum.

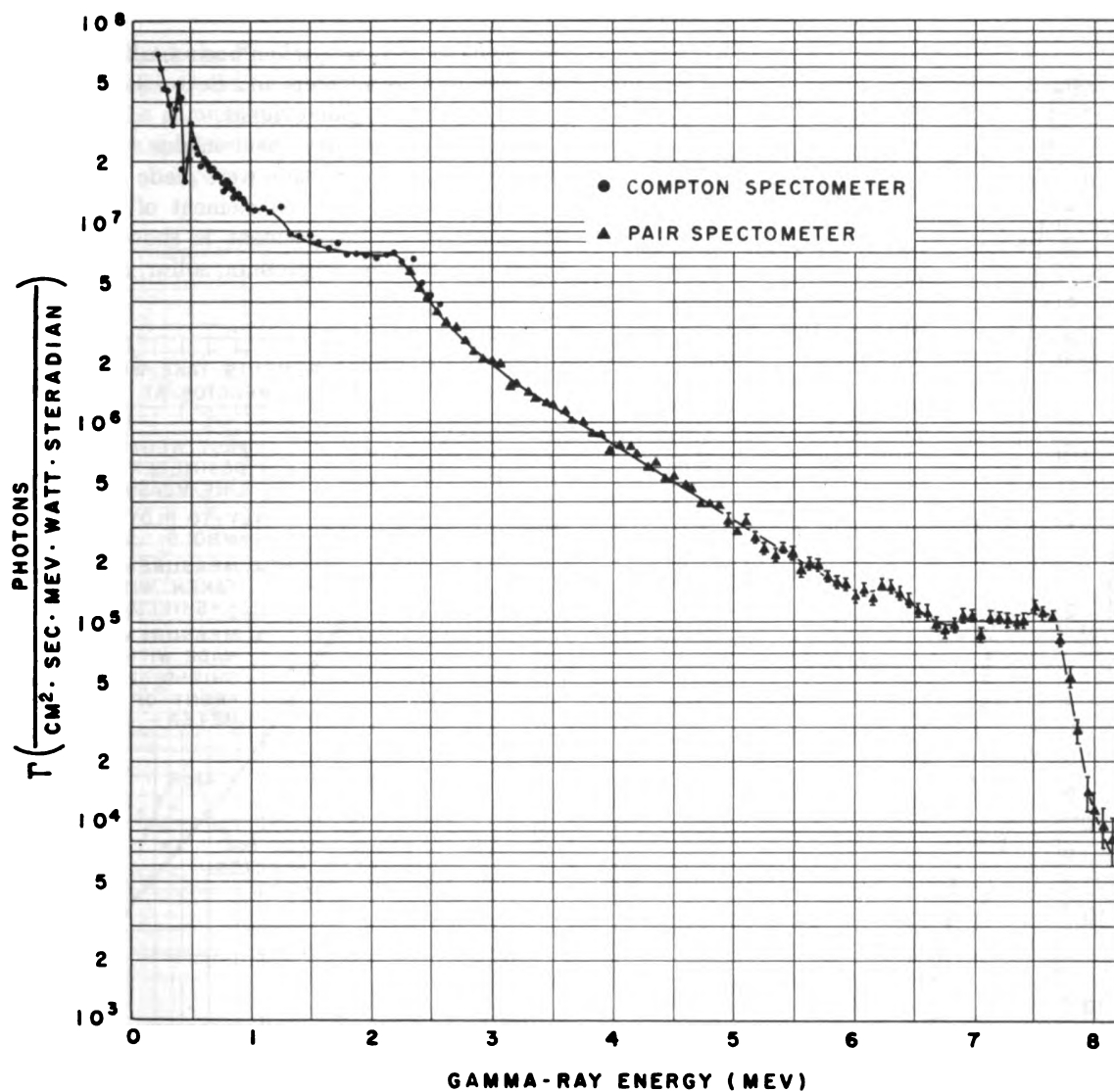


Fig. 2-26 Gamma-ray spectrum of bulk-shielding reactor.

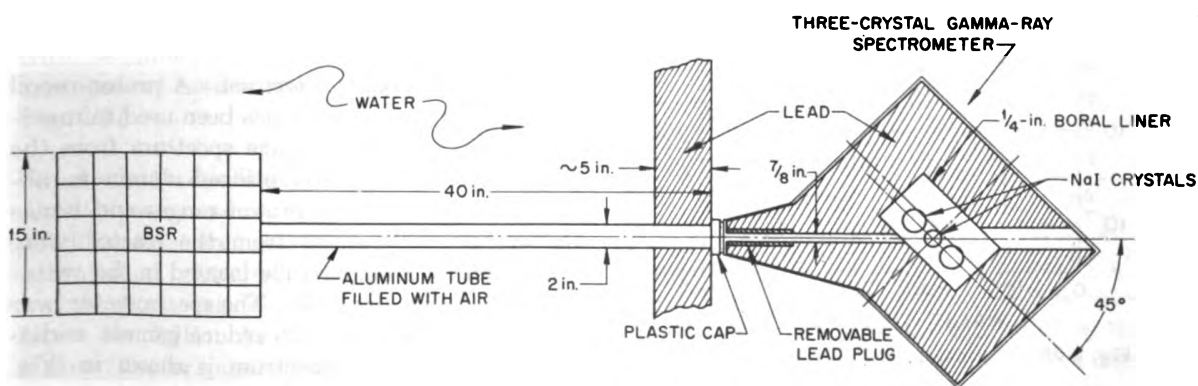


Fig. 2-27 Experimental arrangement for measurement of gamma-ray spectrum.

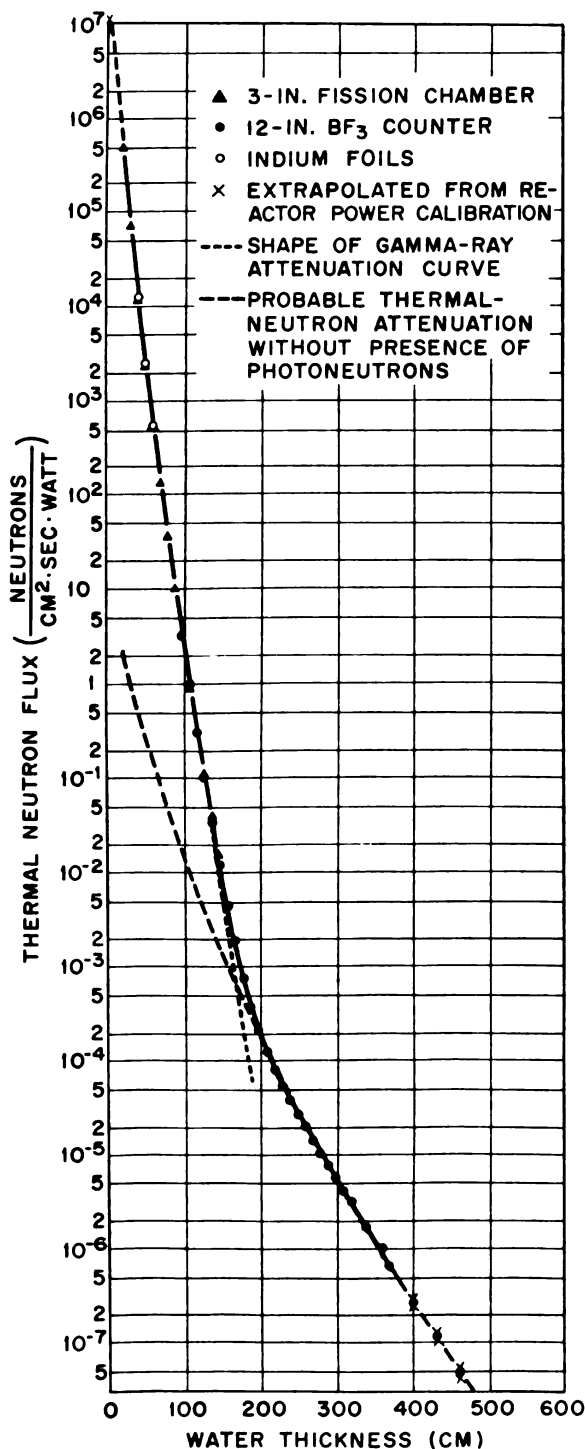


Fig. 2-28 Thermal-neutron flux attenuation.

mal-neutron traverses, taken horizontally through the BSR, for both water- and BeO-reflected reactors. Peaking is again evident.

Thermal to high-energy neutron flux. Measurements using radioactivants were made at various positions in a special fuel element of the BSR. The position of this element is shown in Fig. 2-18d. Aluminum, magnesium, sulfur, and cobalt

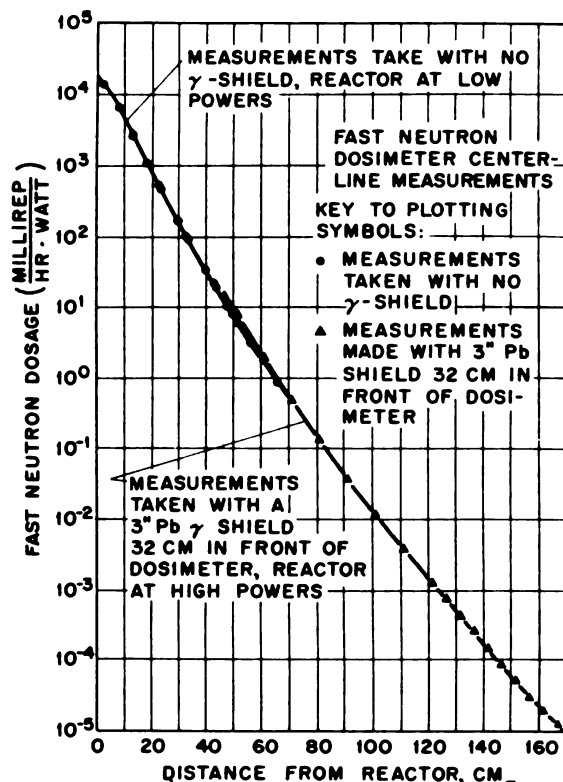


Fig. 2-29 Fast-neutron dose attenuation.

were the radioactivants used. The pertinent reactions together with the resulting data are portrayed in Fig. 2-24.

Fast-neutron leakage spectrum. A proton-recoil fast-neutron spectrometer has been used to measure the fast-neutron leakage spectrum from the BSR. The spectrometer utilized aluminum absorbers to measure the proton ranges and hence the energies. Neutrons from the reactor were collimated by a hollow tube located in the water of the pool near the BSR. The spectrometer was housed in a lead shield to reduce gamma radiation. The resulting spectrum is shown in Fig. 2-25.

Gamma Leakage Spectrum. A multiple-crystal scintillation spectrometer has been used to measure the gamma-ray leakage spectrum of the BSR. This spectrometer utilized both the Compton effect and pair production to give the true

power variation for such runs was between 0.1 watt and 10^6 watts. This large variation and the use of detecting instruments of varying sensitivities permitted the measurements of attenuations as large as 10^{13}

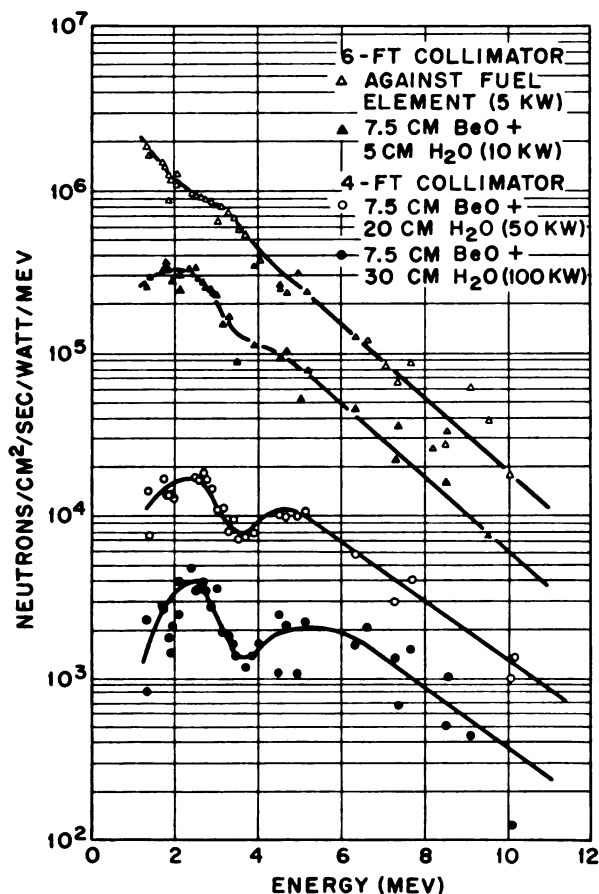


Fig. 2-30 Fast-neutron spectrum through various thicknesses of water.

gamma-ray energy spectrum. The resulting spectrum is shown in Fig. 2-26. The experimental arrangement used at the reactor is illustrated in Fig. 2-27.

Neutron Attenuation in the Water Shield. *Reactor powers and instrument sensitivities.* For all shielding measurements of neutron and gamma attenuation, detectors were positioned in the water of the reactor pool at various distances from the BSR. As the attenuation increased, higher reactor powers were employed to maintain reasonable counting rates. The maximum

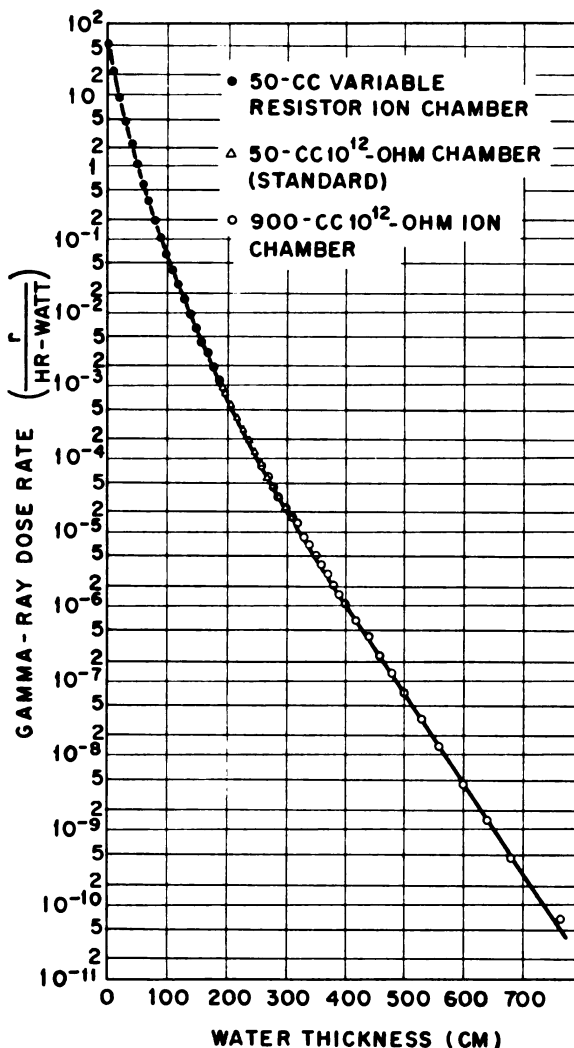


Fig. 2-31 Gamma-ray dose attenuation.

Thermal neutrons. The thermal-neutron curve is plotted in Fig. 2-28. The dashed curve has the same shape as the gamma-ray-dose attenuation curve shown in Fig. 2-31. The similarity in slopes is due to the fact that essentially all neutrons beyond 200 cm are photoneutrons, produced by the gamma rays in the deuterium of the pool water. The difference curve, shown as dotted in Fig. 2-28, represents the probable thermal-neutron

attenuation. This attenuation is exponential with a relaxation length (reduction to $1/e$) which varies between 6 and 9 cm over a reduction of 10^{10} .

Three different detectors were utilized for thermal-neutron flux measurements. Close to the reactor a 3-in.-diameter single-plate U^{235} fission chamber was used. This instrument has little gamma-ray background response and the center of detection is well known. At greater distances, where more sensitivity was required, a large BF_3 proportional counter (12½ in. long and 2 in. in diameter) was used. A conventional amplifier and scaler served for both the counter and chamber.

Absolute calibration of the thermal-neutron flux was provided by exposure of indium foils with and without cadmium covers. The difference of such activations is proportional to the thermal flux. The foil calibrations were made in the ORNL standard pile with due correction being made for the change from graphite to water.

Fast-neutron dose. A fast-neutron dosimeter was used to measure the attenuation of fast-neutron dose in the reactor pool. For measurements far from the reactor, a 3-in. lead shield was placed between the reactor and detector to reduce the gamma-ray background. The results are shown in Fig. 2-29.

Fast-neutron spectrum. Neutron spectra could be measured only through a relatively small thickness of water because of problems of intensity and gamma-ray background. The results are shown in Fig. 2-30.

Gamma Attenuation in the Water Shield.

Dose. The results of gamma-ray dose measurements are shown in Fig. 2-31. The gamma-ray dose was measured with a set of ion chambers utilizing the Bragg-Gray principle. For such chambers the gamma-ray dose is proportional to the ionization current flowing through the grid resistor of an electrometer. The null condition is determined when the voltage across the grid resistor is just balanced by a known bucking voltage.

Three ion chambers with different sensitivities were used in order to cover a large dose range. The sizes were 50-cm³ volume, 10^{12} -ohm resistor; 50-cm³ volume, resistor variable from 10^6 to 10^{11} ohms; 900-cm³ volume, 10^{12} -ohm resistor.

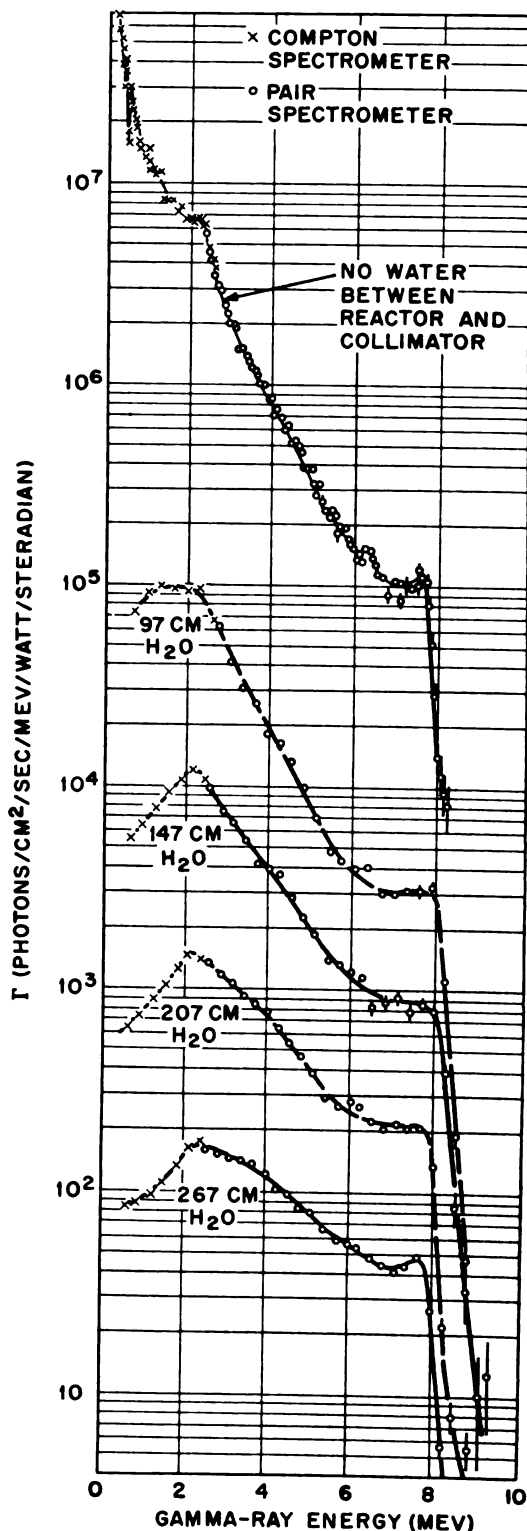


Fig. 2-32 Gamma spectra and attenuation

A check of the absolute calibration was made for one of the chambers with a radium source previously calibrated by the Bureau of Standards. Agreement to within 3 per cent was obtained for favorable geometries.

Spectrum. Gamma-ray spectra, as measured through various thicknesses of water, are shown in Fig. 2-32. The same spectrometer as used for the gamma leakage spectrum (Fig. 2-27) was employed. In all cases only the outward-directed radiation was examined.

EXPERIMENTAL METHODS

Water-temperature Rise. The temperature rise of water passing through the BSR was measured with thermocouples at various power levels. The temperature differences between the top and bottom of the reactor are shown in Fig. 2-33.

Partial Reflectors. Several reflectors were run against the north face of the reactor, using the same loadings as shown in Fig. 2-18a and b. The results are tabulated in Table 2-2. The graphite

Table 2-2 Effect of Partial Reflectors on Reactivity

Reflector *		Reactivity change, % $\Delta k/k$	
Material	Thickness	Water-reflected reactor	BeO-reflected reactor (4 sides)
Boral.....	¼ in.	-1.36 †
Graphite....	1 ft	+0.68
Lead.....	1⅞ in.	+0.38	

* All reflectors extended well beyond the reactor face.

† The boral sheet was held away from the bottom of the reactor face about ¼ in. by the structure supporting the reactor. (Boral is a mixture of B_4C and aluminum, clad with aluminum.)

and lead slabs were mounted on concrete blocks in the bottom of the pool, and the reactor was rolled cautiously against them for the necessary measurements. The boral sheet was hung from above, but the measurements with it were made essentially in the same way as for other reflectors and beam holes.

Duct Effects. Four different ducts (beam holes) were fabricated for these tests. Each duct was sealed after filling with air or saturated boric acid solution, or it was provided with an air line to maintain pressure on the duct during the test.

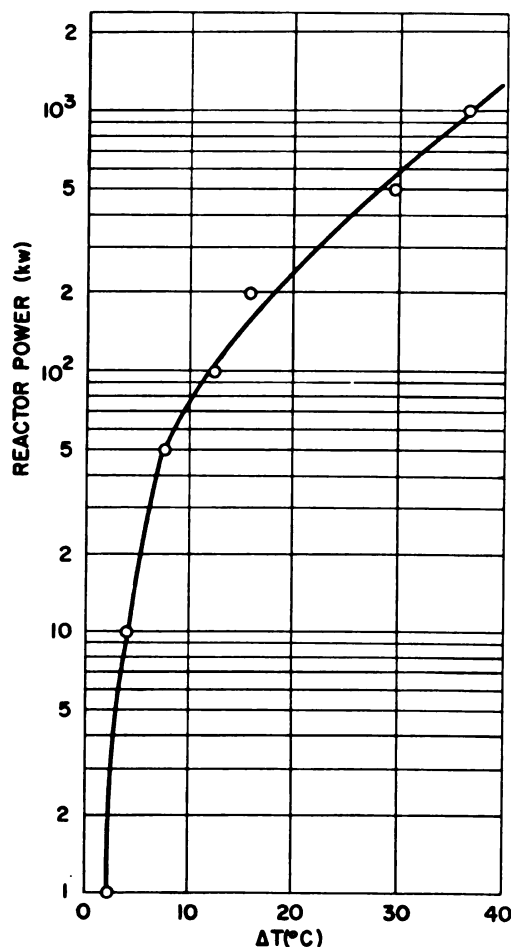


Fig. 2-33 Water-temperature rise as a function of power.

Since the BSR is located in a pool of water, it was necessary to devise a method of remotely positioning each duct against the reactor so that it could be held rigidly while the test was being run. For this purpose, the pit in the large pool was filled with concrete blocks, and ropes were run through the rings in the blocks to each side of the pool, at least four ropes being secured around each duct.

With one exception, each duct was positioned against the north face of the reactor so that the center of the face of the duct coincided with the

center line of the reactor. The reactor was then rolled away from the duct and brought to a low power level. The positions of all three reactor control rods were noted by measurements with a meter stick. The regulating rod was initially positioned so that the reactivity would not be excessive when the reactor and duct were brought together. The reactor was then cautiously rolled against the stationary duct, allowing the servo system to maintain the power level. The new position of the regulating rod was noted and the reactor was shut down. This procedure was repeated with each of the ducts.

The change in reactivity due to the ducts was greater for the beryllium oxide-reflected loading than had been expected. The explanation may lie in an apparent inefficiency of the beryllium oxide reflector elements, the condition of which was not known. The elements are being recanned, but a simple rod calibration by the distributed-poison method is no longer practical because of the large activity of the fuel elements.

The reactivity effect of ducts may be related to their cross-sectional area by the expression

$$\Delta k/k = CD^2$$

where D = diameter of duct, in.

C = constant for a particular loading

Values of C for each duct are shown in Table 2-3. The results for the 6- and 8-in. ducts are quite consistent, yielding an average value for C of 0.009 ($\% \Delta k/k$)/in.² for a water-reflected reactor and 0.007 ($\% \Delta k/k$)/in.² for a beryllium oxide-reflected reactor. The value of the constants for the 17½-in. duct was low because this duct was larger than the reactor.

Radiation Levels and Induced Activities. Figure 2-34 shows typical gamma-ray dose levels at a 100-kw power level with 15½ ft of water above the active lattice. At higher power levels the pumping action of the BSR causes hot water to rise rather rapidly to the surface of the reactor

Table 2-3 Beam-hole Experiments with the BSR

Duct		Duct filler	Reactivity change, $\% \Delta k/k$		$C = (\% \Delta k/k)/D^2$	
OD, in.	Length, in.		Water-reflected reactor	BeO-reflected reactor	Water-reflected reactor	BeO-reflected reactor
6	47¾	Air	-0.36 *	-0.25 *	0.010	0.007
6	47¾	Borated water	-0.245 *		
8½	47¾	Air	-0.55	-0.47	0.009	0.007
17¾	36¾	Air	-1.4	-1.12 †	0.004	0.0035
4¼	44	Air	-0.14 ‡	0.007	
6	12	Air	-0.26	0.007
6	12	Air	-0.10 ¶		
6	47¾	Air	-0.01 §		

* This duct was held approximately ½ in. away from the reactor face by a weld on the face of the duct.

† This duct was larger than the average portion of the lattice.

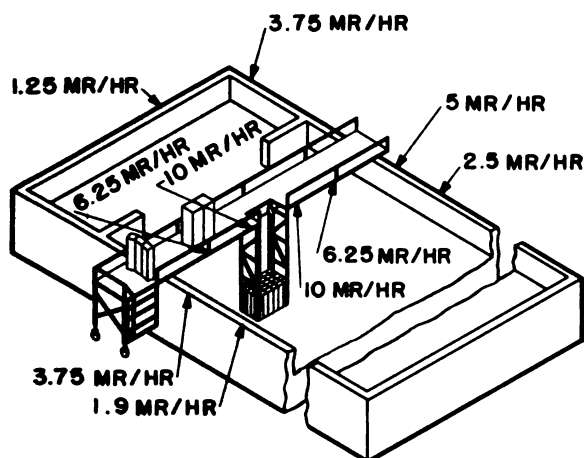
‡ This duct had a ½-in.-thick wall and ½-in.-thick end, while the other ducts had ¼-in.-thick walls and ends.

¶ This measurement was made with the duct at the edge of the north face; the face of the duct was against the reflector elements in lattice positions 2 and 3.

§ This reactivity change results from placing the duct against the boron sheet.

pool. Consequently the 7-sec N^{16} activity induced in the water by the $O^{16}(n,p)N^{16}$ reaction has not yet decayed when the water reaches the surface. The 6-Mev gamma ray from this reaction predominates at high power levels if the direct penetration of gamma radiation is held to

For 1-megawatt operation the water thickness above the reactor was increased to 17 ft. Furthermore, water was prevented from rising rapidly to the surface by a jet diffuser (see Fig. 2-35). At a point on the surface of the water just above the reactor, the gamma-ray dose at 1-megawatt



100-KW (NOMINAL) REACTOR POWER
17 FT. WATER ABOVE REACTOR

Fig. 2-34 Radiation levels around pool at 100 kw nominal power.

tolerance levels. Table 2-4 shows a comparison of the two radiation sources at various power levels. The measurements were made without a jet diffuser.

Table 2-4 Gamma Radiation at Water Surface Directly above Reactor

Reactor power, kw	Measured gamma-ray dose, mr/hr *	Direct penetration gamma-ray dose, mr/hr †
0	1	
1	1	
10	1	
50	3.8	
100	9.3	14
200	49	28
500	303	70
1000	1200	140

* Including N^{16} activity.

† Calculated assuming that direct penetration from the reactor is the only gamma-ray source.

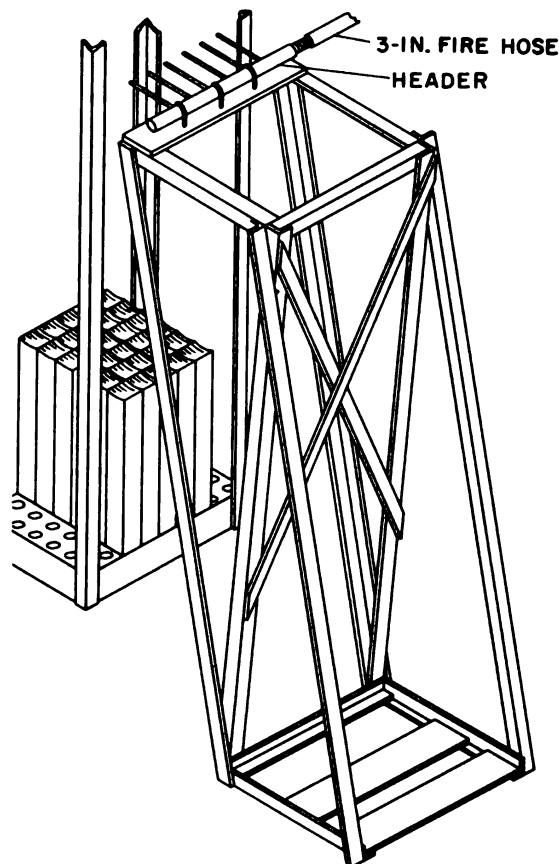


Fig. 2-35 Bulk-annealing reactor with jet diffuser in position.

power was reduced from about 0.8 r/hr to less than 0.1 r/hr by the jet diffuser. At the reactor control panel the gamma-ray dose is near tolerance.

Long-lived activities induced in impurities in the pool water make no measurable contribution to the gamma-ray level around the reactor pool.

Control-rod Calibrations. The control rods of the BSR have been calibrated in three reactor loadings: those shown in Figs. 2-18a and b, and in an additional 5×6 water-reflected array. The control rods used for the latter loading consisted

of two cadmium-lead safety rods, each having an effect of about 1.3 per cent $\Delta k/k$, and a cadmium-lead regulating rod with a rating of about 0.6 per cent $\Delta k/k$ over the last 12 in. of travel.

The control rods used for the configuration shown in Fig. 2-18b consisted of two safety rods containing B_4C powder as the neutron-absorbing material and a water-filled hollow type 347 stainless-steel regulating rod with a 0.087-in.-thick wall. The calibration of these rods is described below.

Inhour calibration method. The reactor was made critical at a power level of about 10 watts and was allowed to stabilize for about 5 min; the positions of all rods were noted. With the safety rods stationary, the regulating rod was then quickly withdrawn to a predetermined position and the rate of increase of the flux level (i.e., the reactor period) between power levels of 100 watts and 1 kw was determined.

It was assumed that the effect of the transient terms on the reactor period would damp out between 10 and 100 watts and that there would be no temperature effect at 1 kw. The period is related to the change in reactivity ($\Delta k/k$) by the inhour equation. Data for the loading shown in Fig. 2-18a are plotted in Fig. 2-36 as an integral curve. In this loading, the stainless-steel regulating rod has a rating of about 0.73 per cent $\Delta k/k$.

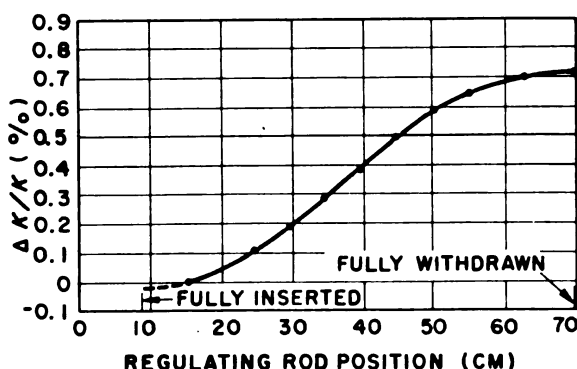


Fig. 2-36 Integral curve for stainless-steel-regulating-rod inhour calibration.

Each of the safety rods in the loading (Fig. 2-18a) was then calibrated against the regulating rod. The reactor was brought to a stable power level of something less than 1 watt and placed under servo control; then the positions of all three rods were noted. The position of one of the safety

rods was changed by a known amount, the other safety rod was left fixed, and the new position of the regulating rod required to maintain constant power was noted. This procedure was repeated until the entire available length of each rod had been covered.

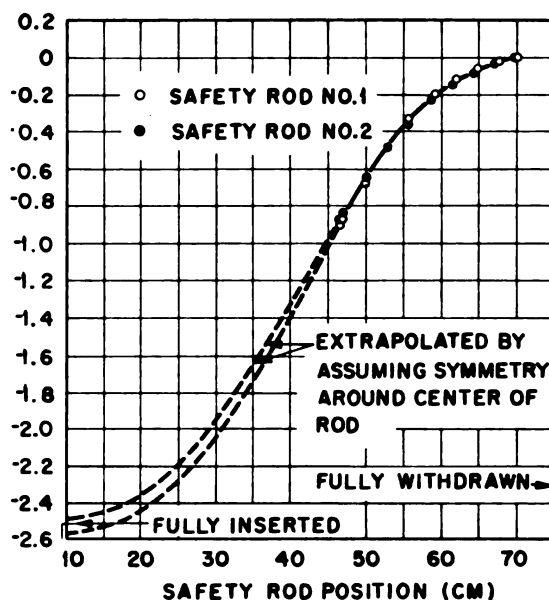


Fig. 2-37 Integral curves for boron carbide safety rods.

Since there was insufficient excess reactivity in the loading to permit the reactor to remain critical with only one safety rod inserted and the regulating rod fully withdrawn, the calibration curves for both safety rods were extrapolated from about the 27-cm position to the fully inserted position by assuming symmetry around the midpoint of the rod. The calibration curves for these rods were also plotted as integral curves to show the effect of insertion of the rod from the fully withdrawn position. Extrapolation of the curves to the fully inserted position indicated a total worth of about 2.5 per cent $\Delta k/k$ for each safety rod in this loading. Figure 2-37 shows these data.

Distributed-poison Calibration Method.

Since the effect of photoneutrons from beryllium makes rod calibrations by the usual inhour formula questionable for a beryllium-reflected loading, the regulating rod for the loading shown in Fig. 2-18b was calibrated by means of distributed gold poisoning. Gold in the form of foils was at-

tached to Lucite strips, which were then inserted into the fuel elements of the reactor lattice in such a pattern that the poisoning was approximately uniform. The reactor was brought to a low, stable power level and the position of the regulating rod was measured, both with and without the poison, while the safety-rod positions remained unchanged. Six different amounts of poison were used, all with the safety rods at the same positions.

The reactor cross section was computed as 5900 cm², by considering only the active portion of the lattice. The change in reactivity resulting from the insertion of a known amount of gold could then be calculated from $\Delta\Sigma/\Sigma$ by using a thermal-neutron absorption cross section for gold of 98 barns. Since the presence of the gold should cause no appreciable change in the Fermi age, $\Delta\Sigma/\Sigma$ should be the same as $\Delta k/k$. The calibration curve for the regulating rod was then plotted as an integral curve showing the change in reactivity from the initial rod position of 17 cm (see Fig. 2-38).

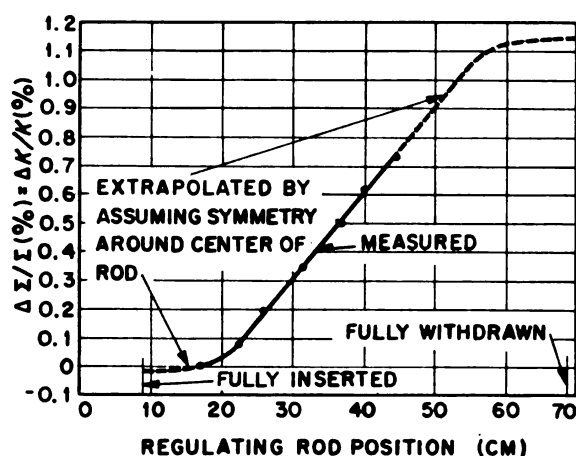


Fig. 2-38 Integral curve for stainless-steel regulating rod, utilizing a distributed poison calibration.

It was necessary to extrapolate the upper portion of the curve above the measured points by assuming symmetry around the point at which the rod is half withdrawn from the lattice. This procedure indicated a total worth of regulating rod of approximately 1.2 per cent $\Delta k/k$.

Power-level Determinations. From a series of thermal-neutron flux traverses a complete map-

ping of the thermal-neutron flux throughout many reactor cores has been obtained. This procedure has required the exposure of as many as 600 foils. From the integration of the thermal-neutron flux over the entire reactor the fission rate may be calculated if the amount of fuel and the fission cross section are known.

Finally the power production is determined from the fission rate by using the value for the

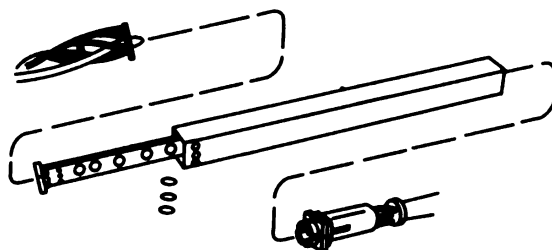


Fig. 2-39 Special fuel assembly for foil exposure, showing foils.

energy produced per fission, as measured for the BSR. This constant, 193 Mev per fission, was obtained by determining the fission rate in one fuel element through exposure of U²³⁵ foils. The special fuel assembly and the foils are illustrated in Fig. 2-39. The core loading used is that shown in Fig. 2-18d. The heat production was determined by insulating the fuel element and observing the temperature rise in water flowing at a constant rate through the element.

Temperature Coefficient Determinations.

The reactor was placed behind the aluminum gate in the small pool, and the large section of the pool was drained. The water in the small pool was heated by injecting steam at the bottom of the pool at the rate of 10,000 lb/hr. Thermocouples were installed at a number of positions around the reactor.

With the reactor held at a constant power of 100 watts by the servo system, the withdrawal of the regulating rod was observed, as the water temperature increased. At this power no appreciable heat was generated by the reactor itself, and the reactor was assumed to be at uniform temperature throughout. The average of all thermocouple readings was considered to represent the reactor

temperature. The water temperature increased from 17.7 to 56°C during the experiment. The resultant change in radioactivity, as determined from the withdrawal of the calibrated rod, has previously been shown in Fig. 2-19.

OPERATING PROCEDURES

Operating procedures comprise instrument checks, startup, steady operation, reactor reloading, and reactor relocation. There are two kinds of operating personnel: reactor supervisors and reactor operators. These are normally professional personnel, either engineers or scientists. Lists of approved personnel are published.

Instrument and Circuit Checks. Instrument and circuit checks always precede each normal startup. If any instrument or circuit is not functioning properly the reactor is not taken critical. The following is the check list used:

1. Insert key in lock and turn on power. (Key is furnished only to approved operating personnel.)
2. Check gas cylinder gauges to ensure an adequate supply of gas. (Cylinders may be changed during a run without reactor shutdown.)
3. Observe gas flow at reactor console; it must be 0.25 ft³/hr for all chambers.
4. Calibrate log *N* meter in accordance with detailed instructions for that instrument.
5. Check interlocks. Indicator light should be on; if not,
 - a. Push scram button to energize holding relay in the circuit.
 - b. Check to see if bridge monitron power is on and radiation indication is below the scram level.
6. Turn on instrument recorders and see that magnet amplifiers are turned on.
7. Calibrate log count-rate meter: Turn switch to CALIBRATE position and adjust gain until recorder indicates 60 counts/sec. Return switch to USE position.
8. Make sure fission chamber is in the DOWN position and recorder is indicating at least 2 counts/sec. If counting rate is below the required level, investigate neutron-source position.
9. Test scram circuits:
 - a. Check each sigma amplifier by raising the safety rods about 1 in. and inserting a

small probe into the hole marked SCRAM on the sigma amplifiers.

- b. Turn log *N* switch to GROUND.
- c. Raise each rod individually about 1 in. Push scram-test button; period sigma should scram.
- d. Return switch to OPERATE position.
- e. Raise rods about 1 in. again and depress red scram button on console; both rods should drop.

10. Check microammeter: Be sure multiplier is on lowest position and external shunt switch is on position 1.

11. Check servo potentiometer: Set demand helipot for desired scale reading on Brown recorder. (Calibration curve is posted above control.)

Normal Startup. Startups are carried out by one of the reactor operators or supervisors. After the power has been turned on and after the instrument and circuit checks have been made, the reactor is taken critical as follows:

1. An announcement is made over the public-address system that the reactor is being taken critical.
2. Critical experiment data in the reactor log book are checked for the particular loading in use to determine safety-rod positions.
3. The safety rod is *carefully* pulled to the predetermined position while the fission-chamber recorder reading is watched for any indication of criticality.
4. The control rod is then pulled until the fission-chamber count-rate meter indicates a slight period. When the chambers have been withdrawn for megawatt operation, the fission chamber will give the best initial level indication. Above 0.1 watt, the period meter, the log *N* recorder, and the recorder on the microammeter should indicate a power increase. The period is always determined from all three of these instruments, never from just the period meter. A 20- to 30-sec period is sufficiently fast to bring the reactor up to power.
5. After the proper power level is reached the reactor is put under servo control. The fission-chamber count rate is adjusted to about 1 register per sec and is connected to the public-address system. The reactor power is announced on the public-address system.

6. The date, time, experiment number, run number, and scale factor are written on Brown charts and recorded in the reactor log.

Normal Operation. When operation at a steady power level has been achieved, the level is announced over the public-address system, as already noted. The fission-chamber monitor may then be connected with the public-address system and the operator is free to work elsewhere within the facility. Reactor log sheets are filled out at hourly intervals.

Reactor Reloading. A new loading for the BSR is undertaken only when two or more reactor supervisors are present. However, a single reactor supervisor and a reactor operator may reload a known configuration.

Whenever a new loading is required by an experimental arrangement, the reactor frame (unloaded) is positioned. Then, as fuel elements are added, the approach to criticality is carefully determined by the method of subcritical multiplication. The excess reactivity is limited to that required by the operating conditions of the experiment.

Moving the Reactor and Reactor Bridge. The reactor bridge is moved only under the direction of one of the reactor supervisors. The key to the reactor bridge is handled in the same manner as the reactor control key. The reactor is partially unloaded whenever the reactor is moved into a new configuration.

COSTS

A pool-type reactor is one of the least expensive reactors to build and operate. The reactor core is somewhat more expensive than that of the small homogeneous water boiler (Type I). However, a pool-type facility does not require hold-up tanks or a stack, since its fission products are contained.

Cost of Bulk-shielding Facility. Costs, as of the summer of 1950, for building and purchasing the components of the bulk-shielding facility at the Oak Ridge National Laboratory are shown in Table 2-5.

Table 2-5

Reactor essentials:

20 fuel elements, at \$120 each (exclusive of cost of enriched uranium).....	\$ 2,400
Reactor assembly: labor, overhead, and materials (motors, magnets, grid superstructure, etc.).....	28,000
Electronic circuits: labor, overhead, and materials (chambers, circuits, recording instruments, etc.).....	28,000
	<hr/>
	\$ 58,400

Other equipment:

BeO reflector (30 elements).....	\$ 17,000
Servo automatic control.....	2,600
Spares (chambers and electronic equipment) ..	9,000
Health-physics instruments (see Appendix B) ..	7,500
	<hr/>
Total.....	\$ 36,100
Building and pool.....	153,000
	<hr/>
Grand total.....	\$247,500

Costs for a New Facility. The cost of building a pool-type facility varies, depending upon the extent of modifications desired, the proportion of work to be carried out by the using agency, the availability of standard components, and the interest of potential contractors in entering the business of supplying reactor facilities and components. In general, estimates of manufacturing costs tend to run higher than the Oak Ridge figures.

One of the more usual types of modifications which have been planned is to provide access to all or part of the pool wall at the reactor level. When this is done it becomes relatively easy to provide horizontal beam tubes and possibly a thermal column. One possibility is to install the pool on sloping ground, with part of the pool embedded and part exposed. Another way is to install a well-type facility with complete access to the pool wall at the core level.

Another possible modification is to change the control-actuator system and the servo system. In general the newer models will be provided with actuators which give positive action in both directions.

A third modification included in the plans for some facilities is to provide arrangements for positive circulation of water in order to permit extended operation at higher powers, up to 1000 kw.

Plans for accomplishing this usually make use of a funnellike device to be placed under the grid plate so that the water may be pumped down through the fuel plates and away from the pool for processing and return. Equipment outside the pool may include a short-period hold-up tank, water-treatment equipment, and cooling equipment.

As with other types of facilities the cost of modifications and reengineering must be taken into account. For example, the cost for reengineering a pool facility to provide for external cooling and

some modifications of the pool and actuators might run from \$50,000 to \$60,000.

In general one of the reasons that costs for a new facility tend to be somewhat higher than those at Oak Ridge is that such facilities usually incorporate larger buildings and more elaborate supporting operations than the bulk-shielding facility. In fact, for some facilities the cost of the reactor proper represents a rather small proportion of the total cost. Estimated total costs for reactor, pool, building, and all supporting equipment range from about \$250,000 to \$1,000,000.

AQUARIUM REACTOR

As noted earlier, in the discussion of the Oak Ridge bulk-shielding facility (BSF), pool reactor is a name popularly given to a class of reactors which uses a particular type of fuel element and which operates immersed in a relatively large pool of water.

The fact that this type of reactor operates with the core completely visible and accessible makes it ideal for a demonstration reactor, as exemplified by the aquarium reactor shown at the Geneva Conference. In addition, it is extremely safe and simple to operate under demonstration conditions.

This section describes some of the design features of the aquarium reactor. The bulk-shielding reactor (BSR), which is the prototype of the pool-type reactor, has just been described. This section does not attempt a complete description of all design features. The emphasis, rather, is to point up the differences between this reactor and most pool-type research reactors.

DESCRIPTION OF FACILITY

The aquarium demonstrates that setting up a research reactor at the Geneva Conference is a practical, relatively simple, and safe operation.

Because the aquarium reactor was intended as a demonstration type, several design features were selected on the basis of their contribution to a more effective display rather than on the basis of greater research usefulness. These features include the following:

1. Reactor supported from the bottom rather than from the top.
2. Fixed position of the reactor within the tank.
3. Absence of reflector elements.
4. Minimization of excess reactivity in the lattice.
5. Free convective cooling only.
6. Small-diameter tank.
7. Absence of beam holes.

The reactor was designed for a 10-kw power level. Experience with the pool reactor has shown that operation at this power level gives a distinct blue glow with dimmed lights. Also, rea-

sonably steady operation at 10 kw will not result in the activation of fuel elements or components beyond convenient limits for reactor disassembly and disposal.

Safety is a paramount feature of the aquarium reactor and is ensured by the following conditions:

1. A low total integrated power.
2. A minimum of excess reactivity in the lattice as far as is operationally reasonable (0.3 to 0.5 per cent $\Delta k/k$).
3. Extra safety devices: an extra rod, an ion chamber, and a safety circuit.

Reactor and Supporting Structure. A perspective view of the reactor installation for the Geneva Conference is illustrated in Fig. 2-40. The reactor, as shown in this figure, consists of an underground steel tank 10 ft in diameter and 22 ft deep, containing light water and the reactor core. The core, consisting of 25 closely stacked fuel elements mounted on a 5-in.-thick aluminum grid plate, is supported from the bottom of the tank. This method of support is used to provide maximum visibility of the core during operation, although in most research reactors of this type the core is suspended from a movable bridge to facilitate the installation of experimental equipment.

The three safety-control rods shown in Figs. 2-40 and 2-41 are operated by drive mechanisms mounted on a platform at the top of the reactor tank. One of these is connected to a servo system to provide automatic control over minor fluctuations in reactivity. Two similar drive mechanisms are used to adjust the position of two fission chambers which measure the neutron flux in the reactor at very low power levels.

Pool and Water Supply. The reactor tank, containing about 48,000 liters of demineralized light water (13,000 gal), is constructed of 10- to 12-mm steel plate, coated on the inside with paint. A suitable transparent tank cover is provided to prevent objects from falling into the tank and interfering with operation of the reactor.

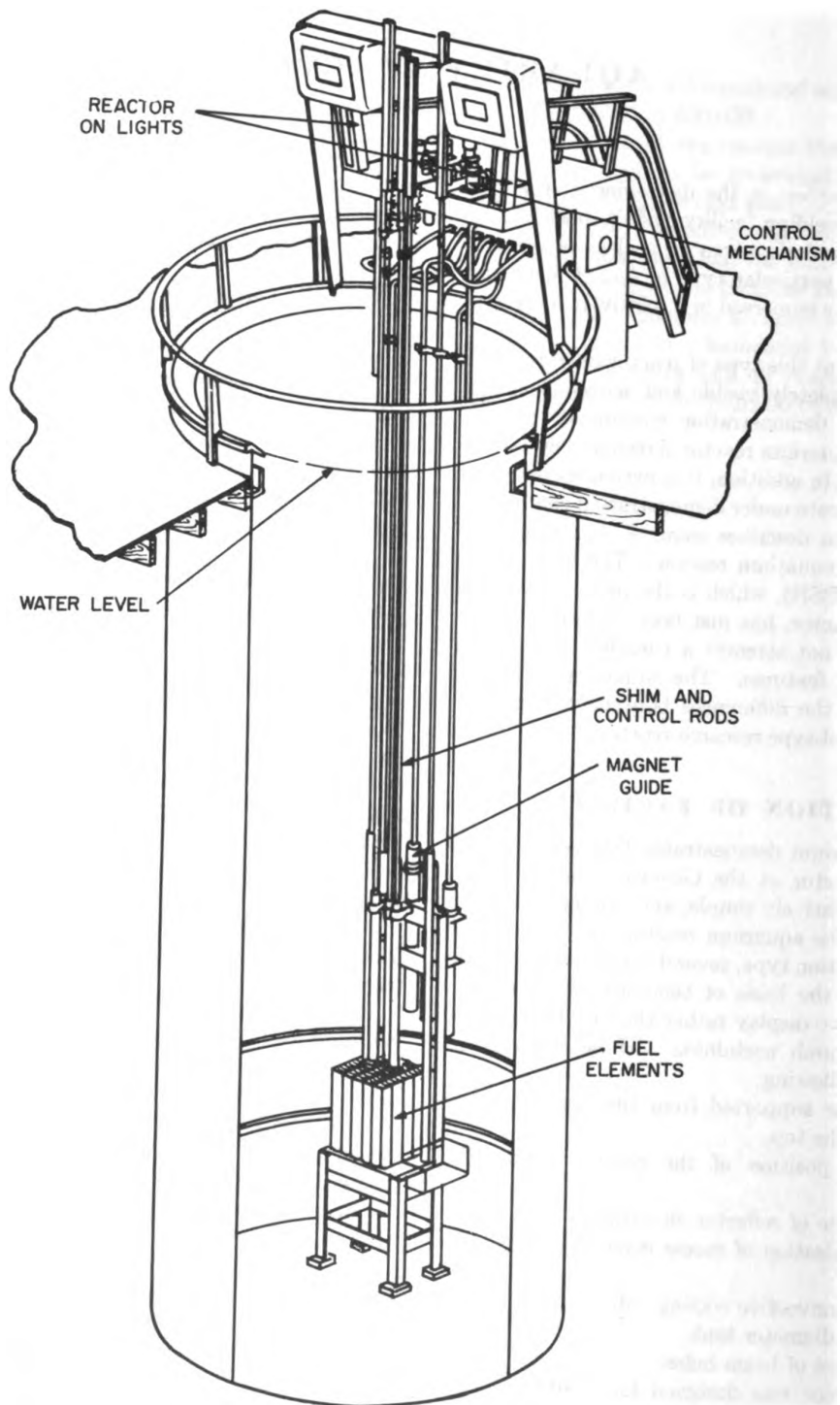


Fig. 2-40 Geneva Conference reactor installation.
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Side and plan views of the reactor tank are shown in Figs. 2-41 and 2-42.

The tank is lighted by means of underwater lights suspended from above. Two-inch IPS con-

adequate space on all sides for observers. The control room, being a vital part of the reactor operation, is given an equally prominent position in the room, with the water-treatment equipment

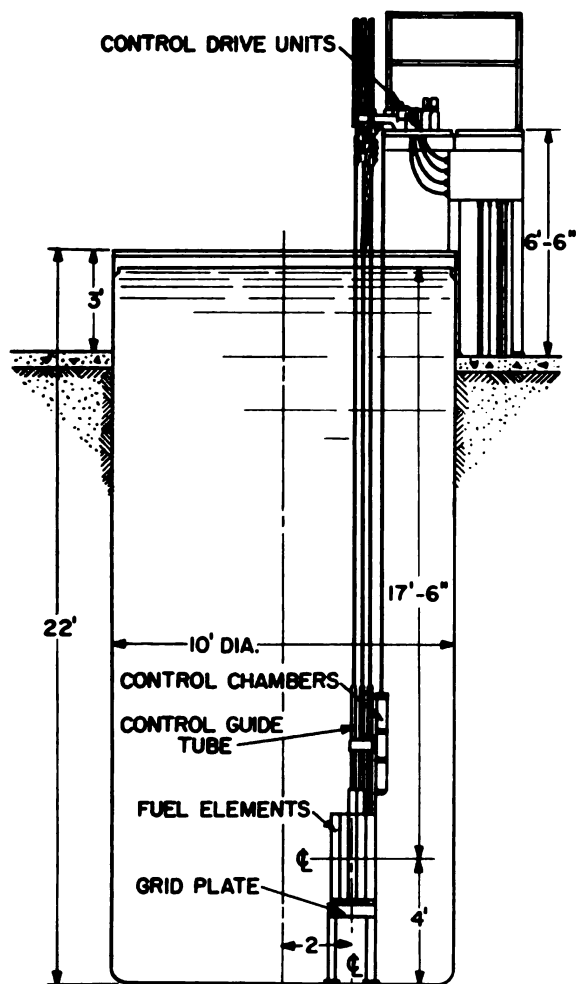


Fig. 2-41 Reactor, elevation.

nections on the tank are provided for charging and circulating water through the tank. Accurately located flat pads are fixed to the bottom of the tank for mounting the stand supporting the grid plate and reactor core.

Building. A floor plan of the reactor building is shown in Fig. 2-43 to indicate the arrangement of the reactor, control room, water-treatment equipment, miscellaneous storage facilities, and corresponding space requirements. As shown, the reactor is centrally located in a main room with

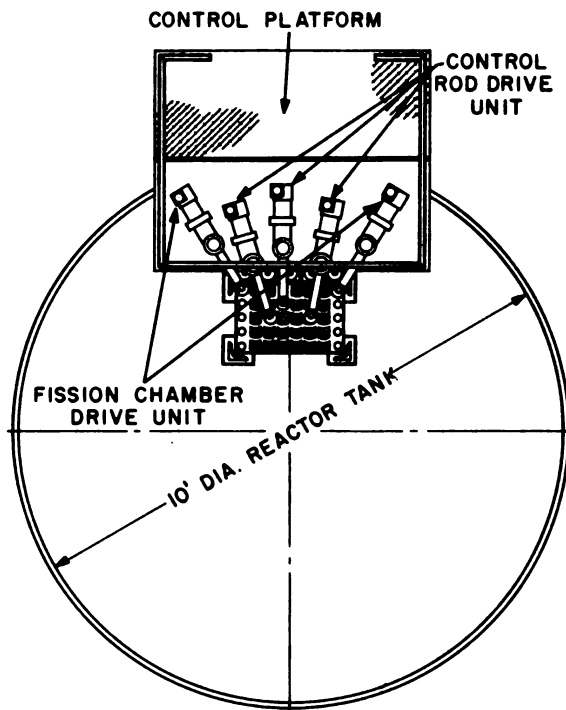


Fig. 2-42 Reactor, plan view.

and other service equipment in side rooms. An outside view of the building is shown in Fig. 2-44.

Radiation Effects and Shielding. The depth of water above the reactor core is 16.5 ft, or 1 ft more than that in the pool reactor. Measurements of the radiation intensity at the surface of the water, 3 ft off the core center line in the pool, give a value of 6 mr/hr at a power level of 100 kw. Since the reactor is normally operated at 10 kw, the estimated radiation intensity is less than one-tenth of permissible tolerance levels.

The induced activity in the steel tank wall adjacent to the reactor core is an important factor in considering the disposition of the tank when the reactor is disassembled. Measurements of the induced gamma activity of reactor materials have been obtained. Data for the saturation activity of typical mild steels are reproduced in Table 2-6. On the basis of these data, the maximum radi-

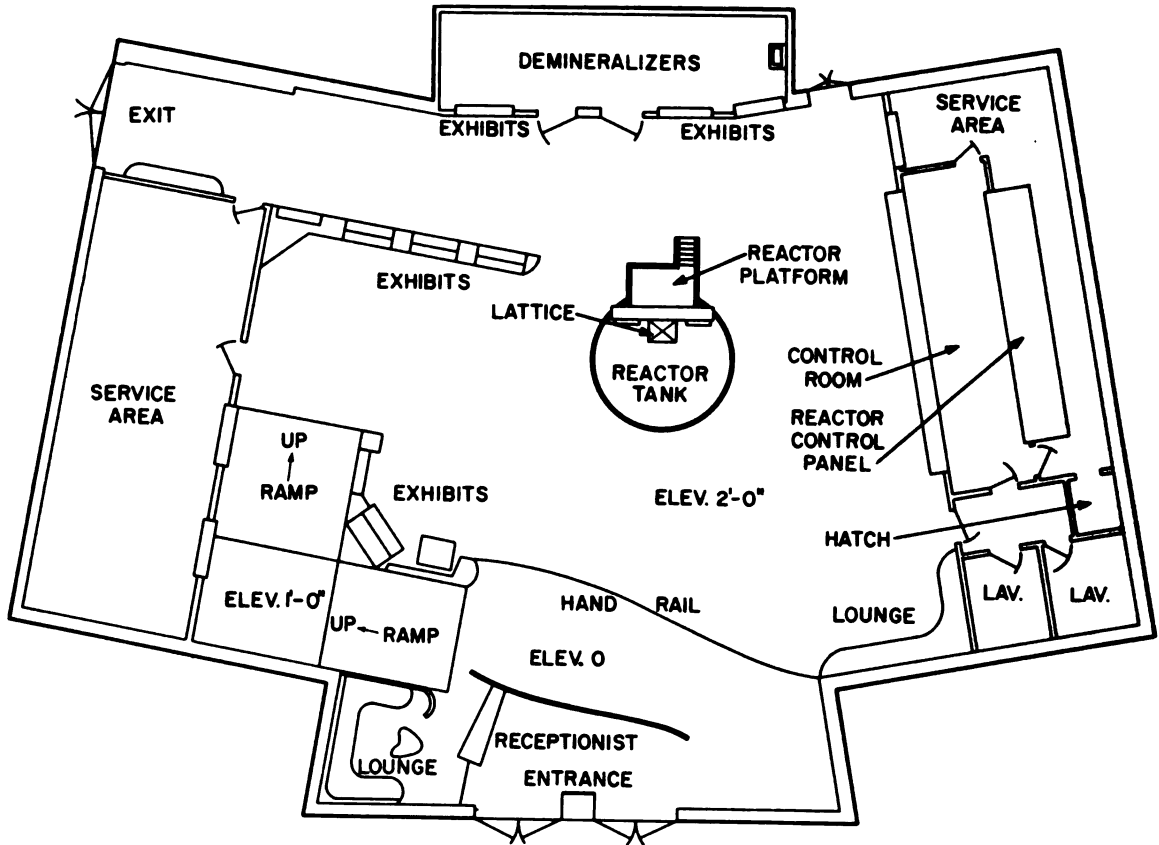


Fig. 2-43 Building, floor plan.

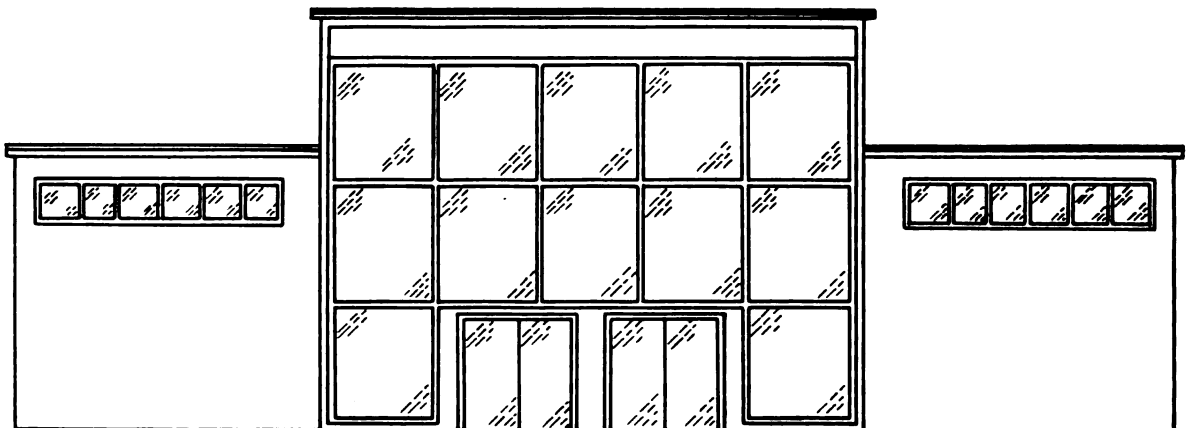


Fig. 2-44 Building, elevation.

tion intensity at the tank wall one day after shutdown will not exceed 0.06 mr/hr. This means that the tank can be handled safely without any special precautions.

Table 2-6 Saturation Activities of 1015 and 1030 Iron Alloys

Half-life	Photons / (sec)(g)(10^{12} flux)
2.5 hr.....	1.8×10^9
45 days.....	3.7×10^7
5.3 years.....	9.0×10^7

The induced activity in the water is important in a consideration of water disposal. Since demineralized water is used, the only appreciable activity will be that due to the escape of aluminum recoil atoms from the fuel-element surfaces. The estimated water activity one day after the end of the 2-week operating period at the Geneva Conference is about 3 dis/sec/ml. At this level the total activity in the tank amounts to 0.004 curie.

The activity of the fuel elements at the end of a 2-week operating period is such that it is advisable to allow the elements to cool about 30 days before shipping them to another location. After 30 days of cooling, each fuel element emits about 8 curies of gamma radiation.

Health-physics Provisions. The usual health-physics provisions and requirements exist for the aquarium reactor. There are fixed instruments and portable instruments. The fixed instruments include the usual monitoring instruments plus air-monitoring instruments and laboratory instruments. Portable instruments include radiation-survey instruments and personnel-survey meters.

For normal monitoring instruments there are four monitrons located as follows:

1. Control drive platform. Two chambers are mounted symmetrically above the reactor tank on the underside of the control platform. The chambers are located in the region of the maximum radiation emerging from the water shield. The chambers are neutron-sensitive and are connected to the reactor control system. If either monitron registers a radiation level of 10 mr/hr or greater, the control rods will be inserted into the lattice, reducing the power level. If both chambers register at least 10 mr/hr, the reactor will be scrammed.

2. Control room. This is an integral chamber-on-chassis installation. It has a neutron-sensitive chamber and is positioned on the reactor side of the control room.

3. Water-processing room. The chamber of this instrument is placed near the columns used for treatment of reactor coolant water.

One instrument, installed in the reactor room, is provided for air monitoring. This instrument provides (1) a demonstration for visitors and (2) a record of air-activity levels throughout the operation.

For laboratory work, one scaler (ORNL instrument department 64-scaler, model 1010) with an end-window GM counter and counter pig is provided. This equipment permits evaluation of the activity levels of various samples, e.g., air, water, smears, etc.

Radiation-survey instruments are of the types in general use at ORNL. In addition to being used for survey purposes they are demonstration items for visitor inspection. They include the following:

1. Cutie pie (portable ionization chamber) (four).
2. GM survey meters (portable Geiger counter) (four).
3. Methane proportional counter (fast-neutron survey meter) (two).
4. Electroscopes. Two provided as a matched pair, one of which has a boron-lined chamber for thermal-neutron measurements.
5. BF_3 counter.
6. Fast air sampler. Filter-collection-type air-sampling device.

THE REACTOR ASSEMBLY

The reactor core consists simply of a stack of 25 pool-type fuel elements, in the arrangement shown in Fig. 2-45, mounted on an aluminum grid plate. The tolerance between the round-end boxes on each fuel element and the sockets in the grid plate is held to close limits to ensure that fuel elements are held rigidly. Three of the fuel elements are designed for insertion of control and safety rods. In addition, one element is designed to hold a neutron source to facilitate control of the reactor during startup. Five ionization chambers placed in cans mounted at the rear face of the

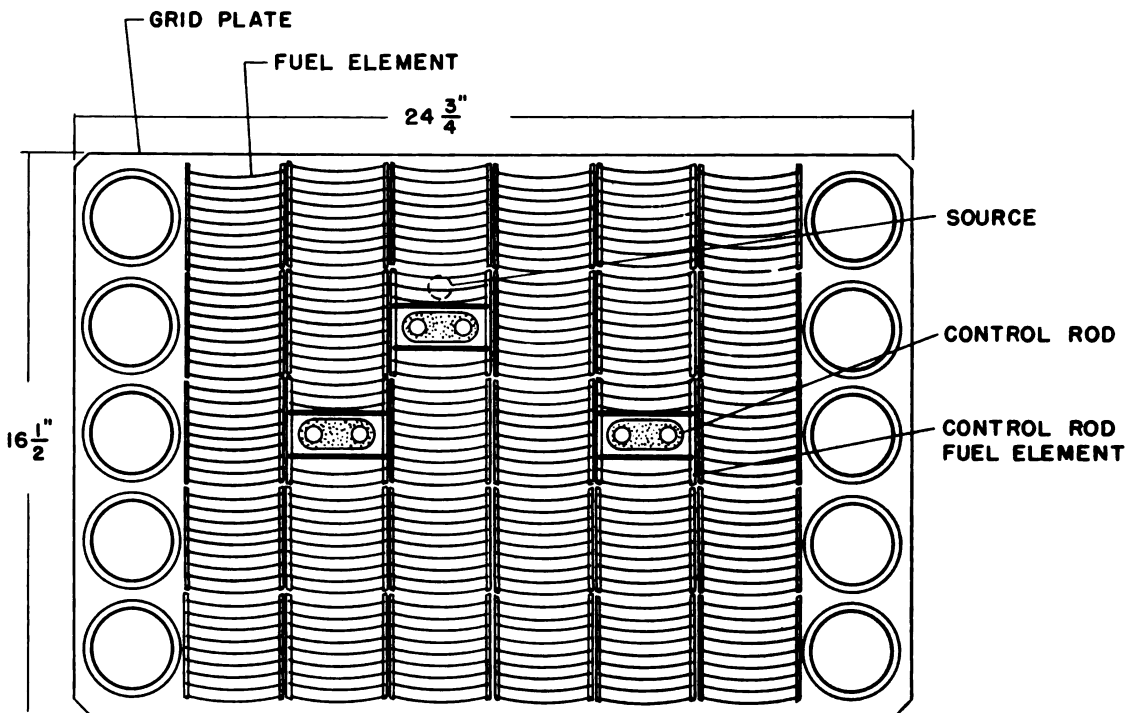


Fig. 2-45 Reactor, section through core.

lattice are required to measure the neutron fluxes over the anticipated range and to activate the safety system for excessively fast periods and for excessive power levels.

The total loading of the reactor is 18.0 kg of uranium, of which 20 per cent, or 3.6 kg, is U^{235} . Details of the design and fabrication of these fuel elements are discussed later.

The mechanisms for adjusting the position of the control rod and for withdrawing and inserting the safety rods are commercially available units. The drive units themselves are accurately positioned above the reactor lattice by means of a steel plate supported by the platform at the top of the tank. Although the total excess reactivity of the core has been limited to 0.3 to 0.5 per cent, which is adequate for the purposes of this demonstration model, each safety rod can control up to 2.0 per cent excess reactivity. Details of the control rods and associated drive mechanisms are given in a later section.

Nuclear Characteristics of Reactor Core. The nuclear characteristics of a conventional pool-type reactor with fuel elements containing highly en-

riched U^{235} and with a water reflector only have been determined experimentally. The nuclear characteristics of such a reactor are compared with those for the aquarium reactor, using 20 per cent U^{235} , in Table 2-7.

Table 2-7 Nuclear Characteristics of Pool-type Cores with ~ 90 Per Cent U^{235} and 20 Per Cent U^{235}

	$\sim 90\% U^{235}$	20% U^{235}
Lattice.....	5 × 6	5 × 5
Number of elements.....	28	25
U^{235} requirement, kg.....	3.5	3.6
Temperature coef, $\Delta k/k$ per °C:		
20°C.....	6×10^{-5}	6×10^{-5}
55°C.....	11×10^{-5}	11×10^{-5}

The loading of the 25 fuel elements in the reactor consists of 20 standard elements, each containing 170 g of U^{235} ; 2 control-rod guide elements, each containing 85 g of U^{235} ; 1 control-rod guide element with built-in neutron source containing 75 g of U^{235} ; and 2 dummy elements containing no

U^{235} . On this basis the total U^{235} in the core amounts to 3645 g.

The excess reactivity required to compensate for possible changes in pool water temperature (20 to 35°C) amounts to 0.1 per cent. An added 0.05 per cent is necessary to offset the small amount of Xe poisoning on successive startups. A total of 0.3 to 0.5 per cent excess reactivity is available for these effects, for miscellaneous experiments, and for starting up the reactor on reasonably short periods.

A study of the kinetic behavior of the reactor upon withdrawal of the regulating rod at its maximum rate indicates that the power level increases on a 30-sec period for addition of 0.2 per cent excess k . Natural convection cooling of unrestricted fuel elements in the pool reactor gives the following quasi-equilibrium power levels for several k introductions into a cold, clean critical reactor: for $\Delta k = 0.1$ per cent, 500 kw; for $\Delta k = 0.2$ per cent, 1500 kw. At this power level, film boiling will occur, requiring proportionately larger increases in k to increase power. The power for 0.3 per cent Δk cannot be reasonably predicted but it will be significantly below the 4000 kw calculated, assuming the absence of film boiling.

Fuel Elements. The reactor is loaded with fissionable U^{235} fuel enriched to the 20 per cent level. The fuel, in the form of uranium dioxide, is incorporated into curved-plate-type fuel components similar in design to the fuel assemblies employed in the MTR, BSR, and other operating research reactors.

The fuel plates for the aquarium reactor consist of enriched UO_2 particles uniformly dispersed and embedded in a matrix of sintered aluminum powder, the resulting material being clad on all sides with wrought 1100-grade aluminum. The core, or fuel-bearing section, of the composite plate is composed of 51 per cent by weight of UO_2 and 49 per cent by weight of aluminum powder. Each standard or 18-plate fuel unit contains 967 g of UO_2 or 170 g of U^{235} material.

Safety Rods and Actuators. Three movable boron carbide shim-safety rods, having nuclear characteristics similar to those used on the BSR, are used to control the reactivity; small excursions of one of them serve as the regulating element.

The core is arranged to accommodate these rods in locations appropriate to the fuel disposition. The rod drive mechanisms are installed at locations corresponding to these selected rod positions.

Electromagnets attached to each of the rod drive mechanisms provide a means of coupling the drive units to the shim-safety rods. The magnets are energized by current supplied by the magnet amplifier sections of composite amplifiers. Switches are provided to detect whether or not the rods are coupled to their magnets. To indicate when the rods are fully inserted in the lattice, seat switches are employed. Means for establishing the positions of the rod drive mechanisms are installed along with upper- and lower-limit switches.

CONTROL AND SAFETY

Reactivity will be controlled by means of shim-safety rods having nuclear characteristics similar to those used in the BSR. Three such rods are installed, each controlling approximately 2 per cent Δk . The rods are suspended from electromagnets from which they may be released to initiate a shutdown. The magnets are driven up and down by electric motors, through appropriate gear trains. The use of synchronous or induction a-c motors ensures that the maximum speed can be no greater than the synchronous speed determined by the power frequency. For the same reason, the power converter (50 to 60 cycles) is of such a nature that it cannot overspeed.

The speed of rod withdrawal, and hence the rate of reactivity addition, are sufficient to permit a 30-sec-period startup, or a total time of not less than 3 min or more than 6 min. The withdrawal rate is well below the maximum safe rate. The desirability of a high counting rate as the reactor approaches criticality, along with other operating considerations, has determined the withdrawal rate stated above.

Since this reactor is designed to be operated with frequent startups and low integrated energy, the neutron flux varies routinely over its entire range. The usual operating power is 10 kw, with short-time operation up to 100 kw. The lowest power to be measured, except during critical experiments, is set by the magnitude of the source and by the multiplication of the lattice when the

rods are fully inserted. Since the shutdown multiplication is fixed by the loading and the rod reactivities, the magnitude of the source will determine the reactor power under shutdown conditions. Operation during startup is always less vulnerable to operator errors and more easily

this reactor makes use of a full complement of three safety channels. On a short-time basis, failure of one safety channel would not require reactor shutdown; therefore, by the use of three safety channels, there is both added protection and an installed spare. Likewise, in the case of

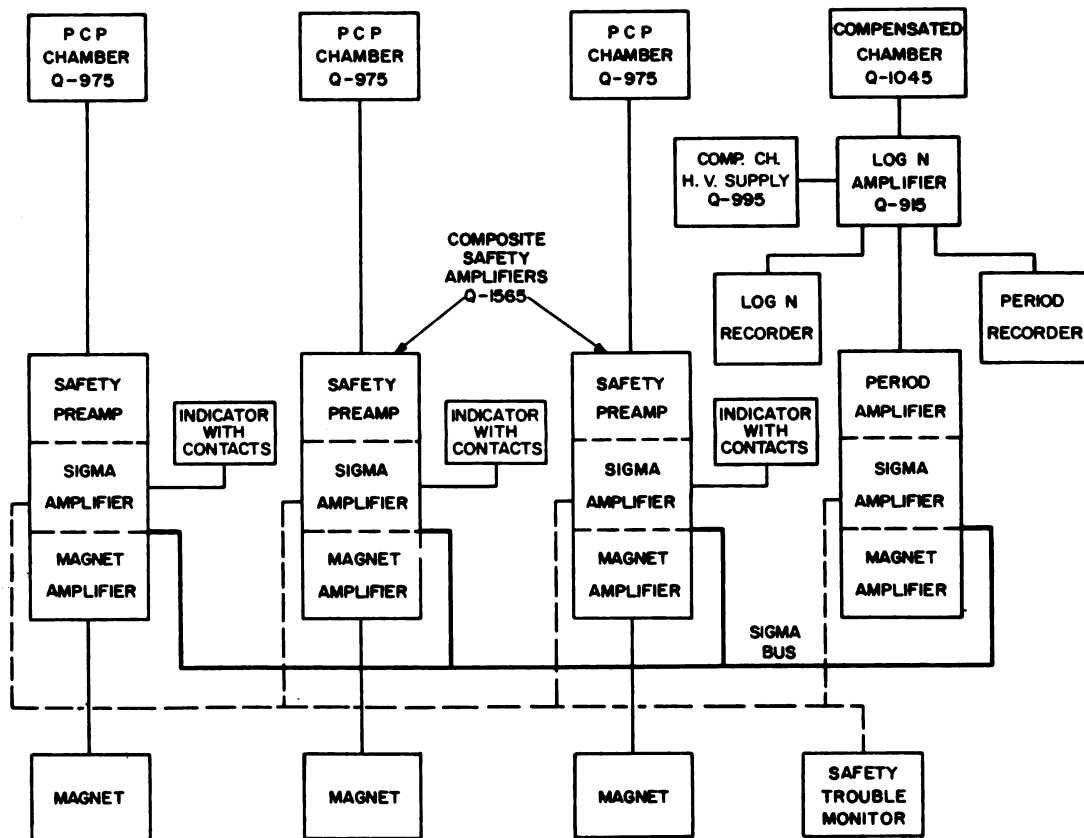


Fig. 2-46 Instrument block diagram, safety channels.

demonstrated when the source is as large as can be arranged. Recent experiments in the BSF give an expected shutdown power of approximately 10 milliwatts.

The reactor control and safety system consists of three neutron-level safety channels (Fig. 2-46), two counting-rate channels (Fig. 2-47), one period channel (Fig. 2-46), and one linear channel with which a servo control is associated (Fig. 2-48). Reactors employing a minimum safety have relied on as few as two safety channels, but because of demonstration conditions,

the counting-rate channels, the additional channel can be considered an installed spare thereby permitting uninterrupted operation in the event of failure in one of the channels. The level safety channels are relied upon to shut the reactor down in the event of operator or control malfunction.

Nuclear information is available as follows:

1. Level safety. Three PCP chambers supply current to three level safety channels (Fig. 2-46). (This is the primary safety device of the reactor.)

2. Period. One compensated chamber supplies current to a period channel. Log N and period are displayed on recorders (Fig. 2-46).

3. Linear. One compensated chamber supplies current to a micromicroammeter (Fig. 2-48). This is a stable d-c amplifier with many current

be identical with that for the shim control. The anticipated rate of reactivity increase for the shim-safeties will be no greater than 0.05 per cent per second, with all rods withdrawing simultaneously.

4. Counting rate. Two fission chambers supply

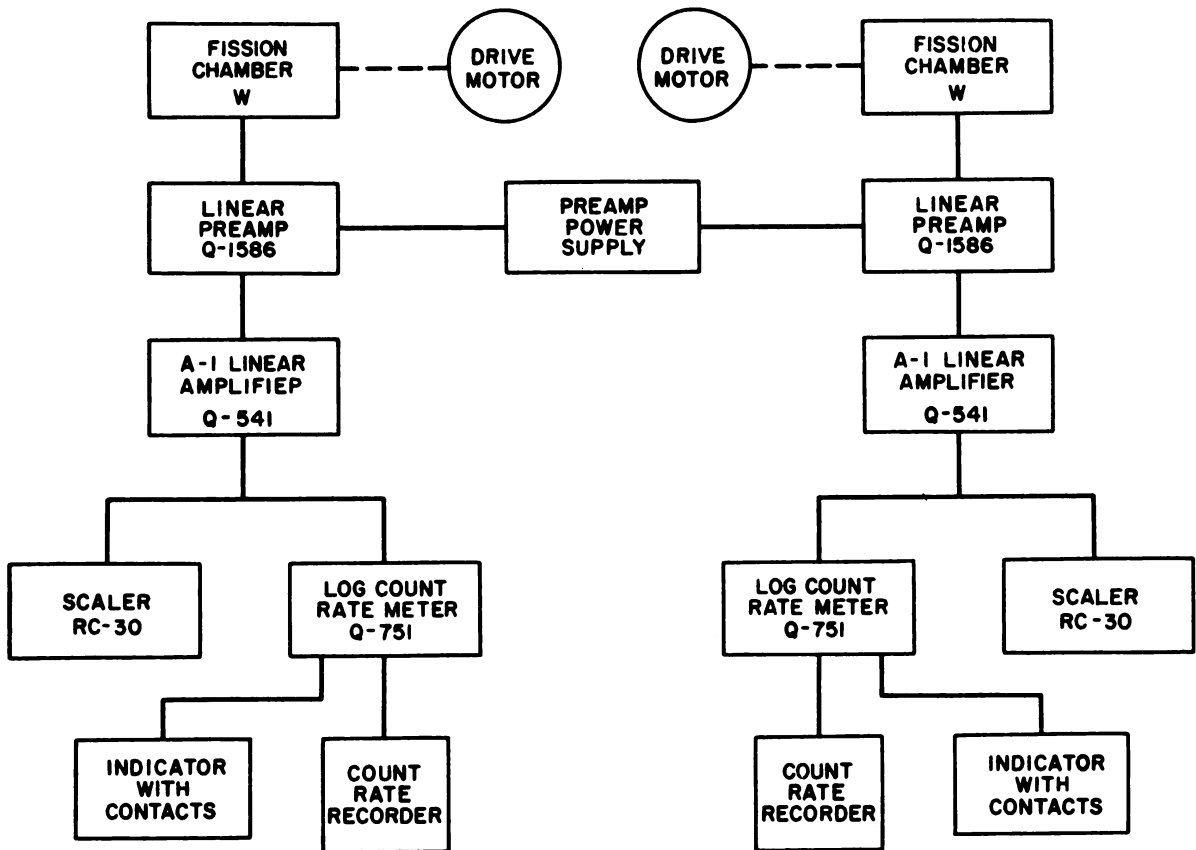


Fig. 2-47 Instrument block diagram, counting-rate channels.

ranges; it is usable at currents as low as 10^{-10} amp and higher than 10^{-4} amp (the current at which the chamber begins to show saturation effects). The micromicroammeter has been provided with means for remotely changing the current range. The output signal from the micromicroammeter operates two recorders, one on the control cubicle and one at the reactor, and also serves as the input signal for a flux servo, which maintains the reactor power constant at a preset level by using small excursions of one of the shim-safety rods as a regulating rod. The reactivity available to the servo control will be limited by the loaded excess k . The rate of rod motion will

pulses to two counting-rate channels (Fig. 2-47). The logarithms of the counting rates are displayed on two recorders with a span of 4 decades. Two additional rod drive mechanisms are used to retract the fission chambers from the lattice as required to keep the instrumentation in range. Period information from the counting-rate channels will be used to supervise rod withdrawal at fluxes below the lower limit of the period channel.

Radiation level at the facility is checked by four monitrons (Fig. 2-49). There is one each for the control room and demineralizer area, and two for the reactor area. The radiation levels indicated by these monitrons will be recorded by a

multipoint recorder located on the instrument panel (Fig. 2-50).

The monitrons are equipped with contacts which give an alarm if the instrument shows a dan-

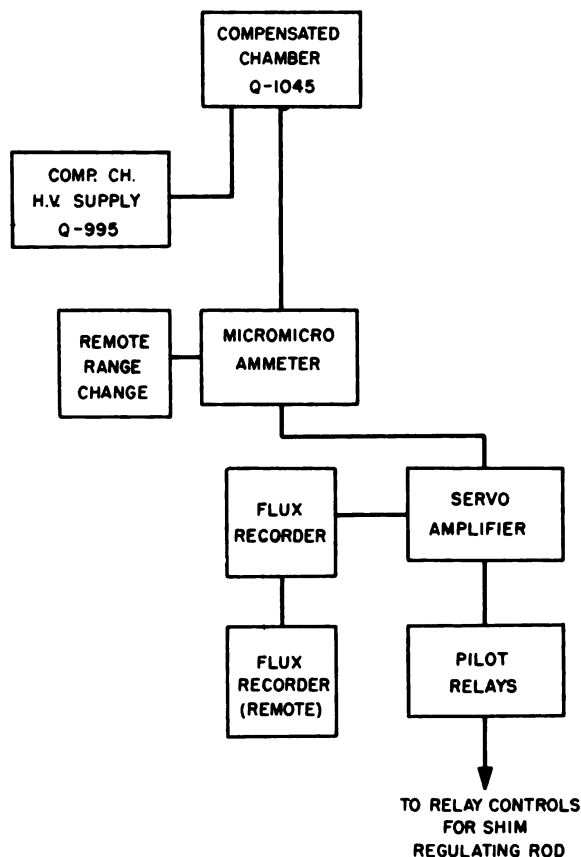


Fig. 2-48 Instrument block diagram, linear and servo channel.

gerously high level or if it is made inoperative. A single alarm from either reactor-area monitron will cause a reverse (motor-driven insertion of all rods); two such alarms occurring simultaneously will initiate a scram (dropping of all rods). A *constant air monitor* is incorporated for testing air contamination.

REACTIVITY CALCULATIONS

U²³⁵ Requirements. The critical mass for the aquarium reactor was determined experimentally.

The significant changes from the pool reactor described in ORNL 1682 are:

1. The substitution of 20 per cent enriched uranium for 90 per cent enriched uranium, which decreases the resonance escape probability and the number of fission neutrons produced per thermal neutron absorbed in the fuel.

2. A reduction of the required excess reactivity to about one-fourth of its value in the pool reactor.

3. A slight reduction in the thickness of the aluminum side plates, which, by decreasing the aluminum-to-water ratio, decreases the fast leakage of neutrons.

Excess Reactivity Requirements. For 10-kw operation, the excess reactivity requirements are:

Operational requirement.....	0.0015
Equilibrium xenon poisoning.....	0.0005
Temperature excursion (20-35°C)...	0.0010
	0.0030

These are calculated as follows:

For changing the operating temperature from 20 to 35°C, a reactivity change of 0.00105 is required (this is taken from Fig. 9B of ORNL 1682, which gives the reactivity change vs. temperature).

The equilibrium xenon reactivity loss is given by the following equation:

$$\frac{\Delta k_{Xe}}{k_{Xe}} = \frac{\gamma \Sigma_f \phi \sigma_{Xe} / \Sigma_T}{\lambda_1 + \sigma_{Xe} \phi}$$

where γ = total yield of xenon per fission = 0.064
 f = macroscopic fission cross section = 0.0500 cm⁻¹

ϕ = average flux (for 10 kw power) = 5.642 × 10¹⁰ neutrons/cm²/sec

Σ_T = total cross section

λ_1 = decay constant for I¹³⁵ = 2.092 × 10⁻⁵ sec⁻¹

σ_{Xe} = averaged xenon cross section = 3.103 × 10⁻¹⁸ cm²

It was assumed that 1 watt requires 3.24 × 10¹⁰ fissions/sec and that the importance factor α is 1.5.

Kinetics. Since for similar reactors the neutron lifetime is inversely proportional to the total thermal-absorption cross section, the lifetime for the aquarium reactor will be 5.17 × 10⁻⁵ sec if

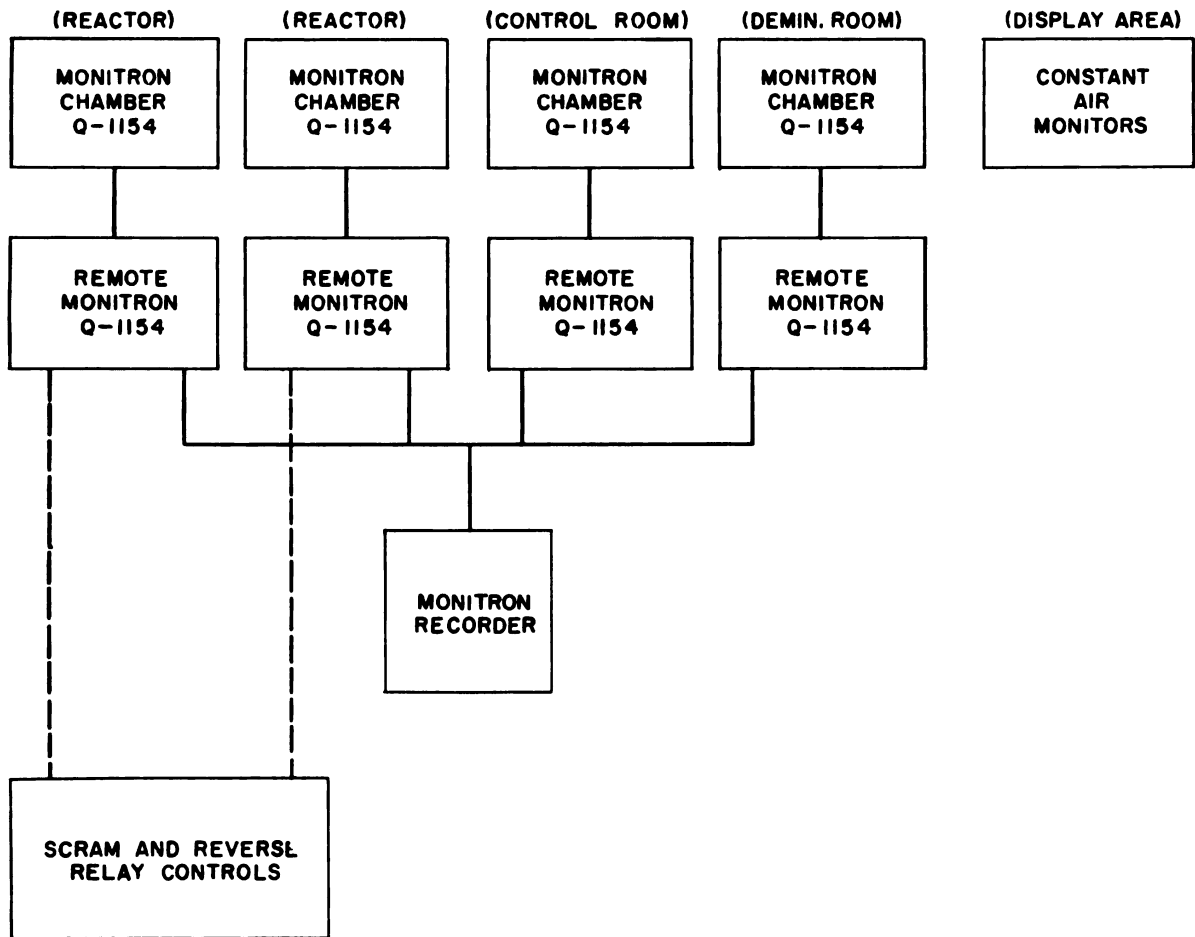


Fig. 2-49 Instrument block diagram, health-physics instruments.

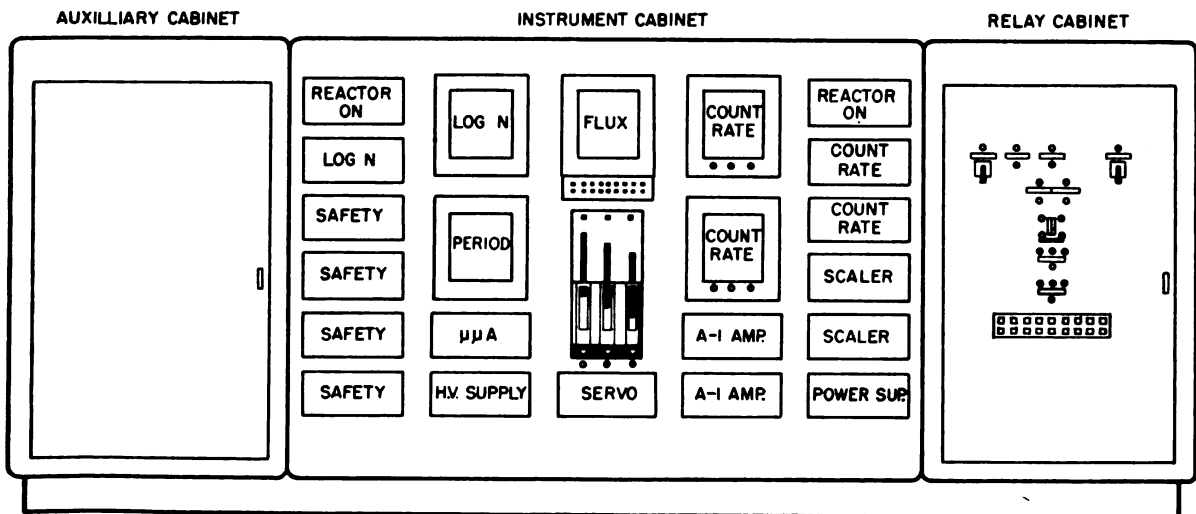


Fig. 2-50 Instrument-control cubicle.

the assumed 5.54×10^{-5} sec lifetime is accepted for the pool reactor.

The usual inhour equation (page 7, ORNL 1682) gives the following stable periods as a function of instantaneous reactivity additions:

Δk_{eff}	Period, sec
0.001	90
0.002	31
0.003	14
0.004	5.9

Quasi-equilibrium Power Levels. For a given change in reactivity the average temperature increase to compensate for this is obtained as above, and the quasi-equilibrium power level is obtained by estimating the temperature change across the reactor (1.5 to 2 times average temperature change), using Fig. 15 of ORNL 1682, which presents power levels as a function of temperature difference.

FUEL REQUIREMENTS AND SPECIFICATIONS

Fuel-loading Requirements. The reactor core has been designed to accommodate 25 fuel components in 5×5 lattice arrangement. The fuel-component requirements for such loading would normally call for 20 standard 170-g units, and other units previously described. Several additional units will be required, however, to limit closely the excess reactivity in the core.

Fuel-element Specifications. The design data on a standard 18-plate fuel unit are summarized in Table 2-8.

Table 2-8

Number of fuel-bearing plates.....	18
Final side-plate thickness, mils.....	110
Water-gap spacing, mils.....	117
Metal-to-water-volume ratio.....	0.63
Core-mixture composition, % by wt.....	51 UO_2 -49 Al
Core-mixture composition, % by vol.....	20 UO_2 -78 Al-2 void
Clad-core-clad-plate thickness, mils ²	17.5-25-17.5
U^{235} distribution per plate, g/cm.....	0.025
U^{235} enrichment factor, %.....	20
U^{235} content per assembly, g.....	170
UO_2 content per assembly, g.....	967
Aluminum content per assembly, kg.....	3.69

Values on the material content refer only to materials contained in the $23\frac{5}{8}$ -in. (60-cm) effective fuel length. The metal-to-water-volume ratio

was calculated on the basis of assigning one-half of the water-gap spacing between adjacent fuel assemblies to the element.

REACTOR SHIELDING

Operation of the reactor at 100 kw is based on experience with the ORNL pool reactor; data are given in Table 2-9 and are plotted in Fig. 2-51.

Table 2-9 Radiation Intensity in Water above Reactor Core

(3 ft from vertical center line)

Depth of water above core, ft	mr/hr
13.0	140
14	55
15	21
16	9
17	4
18	1.8

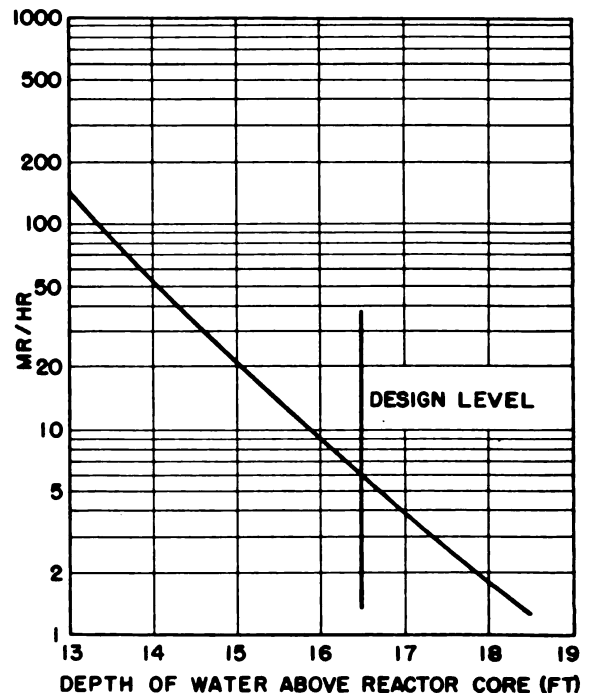


Fig. 2-51 Radiation intensity 3 ft off vertical center line above aquarium reactor (at 100 kw).

INDUCED ACTIVITIES

Induced Activity in Tank Wall. The activity caused by the thermal-neutron irradiation of the

carbon-steel tank has been calculated as a function of the distance between the reactor core and the tank. Neutron fluxes are based on center-line measurements in the ORNL pool reactor (ORNL Drawing 13769), and the resulting activities are based on Bopp's measurements on 1045 steel (as given in Report ORNL 1371). Neglecting the short half-life activity (2.5 hr), the induced activity in the tank wall for the 45-day component is given by the product of

7.25 g Fe/cm² of tank wall ($\frac{3}{8}$ in. thick)
 3.7×10^{-5} photons/sec (g) (unit flux)
 ϕ thermal flux at tank wall
 0.693/45 decay constant, day⁻¹
 8.33 days of operation of reactor
 1.25 Mev/photon
 1/520 mr/hr per Mev/sec
 1/2 (to allow for fact that half of the radiation goes to ground on outside of tank)

$$4.1 \times 10^{-8} \phi \text{ mr/hr}$$

Assuming that 1 beta particle is emitted per photon, the activity in the tank wall is given by the product of

7.25 g Fe/cm²
 3.7×10^{-5} photons/sec (g) (unit flux)
 ϕ thermal flux
 0.693/45 decay constant, day⁻¹
 8.33 days of operation
 0.0066 fractional escape of betas (one-fourth the range)
 1/6.8 betas/cm²/sec per mr/hr

$$3.3 \times 10^{-8} \phi \text{ mr/hr}$$

Thus the total contribution from both betas and gammas is $7.4 \times 10^{-8} \phi$. Values of ϕ are given

Table 2-10

Thickness of water between core and tank wall, ft	ϕ	mr/hr at tank surface
1	7.4×10^8	55
2	5.5×10^8	0.40
3	9.2×10^4	0.007

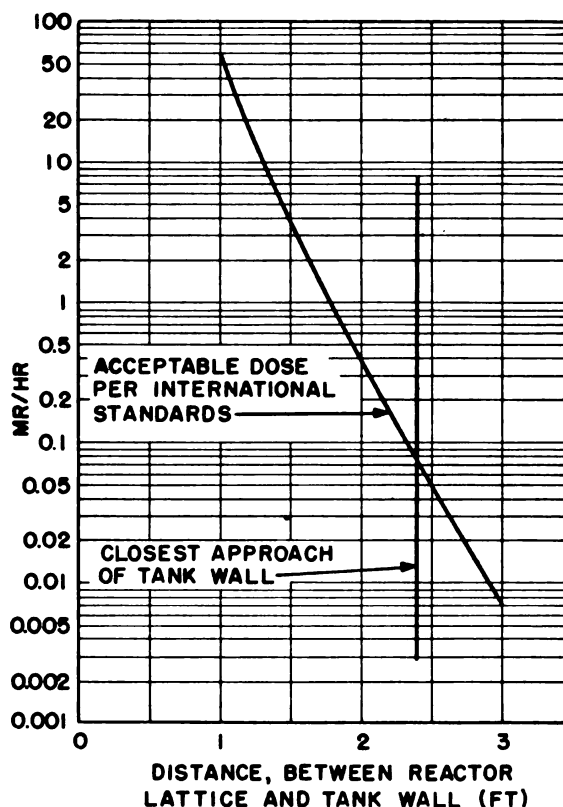


Fig. 2-52 Induced activity of tank wall.

in Table 2-10, and corresponding radiation intensities are plotted in Fig. 2-52.

Radioactivity in Pool Water. Estimates have been made of the activity which may be introduced into the water by recoils from the core. It is to be noted that the tank surface is painted and will not introduce target ions into the water.

The reactions included in the study and the constants employed are shown in Table 2-11, along

Table 2-11 Predicated Radioactivity in Pool from Neutron-reaction Recoils

Reaction	Product half-life	Activation cross section	Range in aluminum, cm	Dis/sec/ml in aquarium water
$\text{Al}^{27} (n, \gamma) \text{Al}^{28}$	2.3 min	0.21 barn	1.2×10^{-6}	33
$\text{Al}^{27} (n, p) \text{Mg}^{27}$	9.5 min	2.8 mb	2.4×10^{-4}	8
$\text{Al}^{27} (n, \alpha) \text{Na}^{24}$	15.1 hr	0.6 mb	2.4×10^{-4}	6
$\text{Na}^{23} (n, \gamma) \text{Na}^{24}$	15.1 hr	0.4 barn	2.4×10^{-4}	6
$\text{Mn}^{55} (n, \gamma) \text{Mn}^{56}$	2.6 hr	13.0 barns	1×10^{-5}	1

with the equilibrium activity resulting from dispersing the radioactive atoms uniformly through the pool.

Thus the radioactivity of the pool water one day after shutdown will be of the order of 3 dis/sec/ml. The total radioactivity of the 48,000 liters will amount to 0.004 curie.

USED FUEL-ELEMENT SHIELDING

Calculations have been made to determine the residual activity of the fuel elements from the reactor. These data are necessary for evaluating the difficulties of handling the assemblies out of water for loading to shipping casks and for determining the shielding thickness and weight of casks for shipment of the irradiated fuel.

In these calculations, it was assumed that the reactor has operated at 10 kw for 12 hr a day for 20 days, thus amassing a total power output of 2400 kwhr. Cooling times of 30 to 150 days were studied. At 30 days the gamma activity is decreased to 1.5×10^{-4} gamma watt per operating

watt and at 150 days' cooling, to 1.5×10^{-5} gamma watt per operating watt. In the shielding calculations, 2.5 and 1.69 Mev La^{140} activity were considered separately, and all other activity was treated as 0.8 Mev. For 30-day cooling, the La^{140} activity is controlling, whereas at 150 days, it has decayed sufficiently to be of secondary importance.

The activity of a single assembly for various cooling times and distances from the source is given in Fig. 2-53.

The results of calculations made to determine the thickness of shielding for a 16-assembly car-

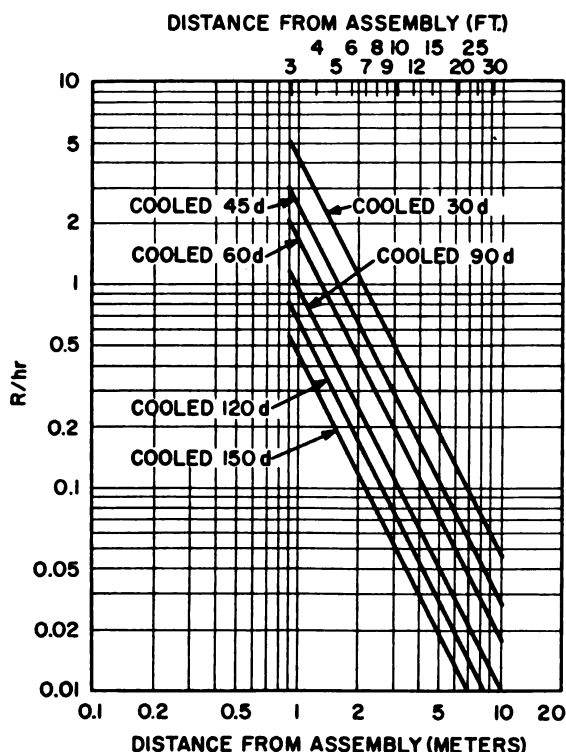


Fig. 2-53 Activity of a single assembly for various cooling times and distances from source.

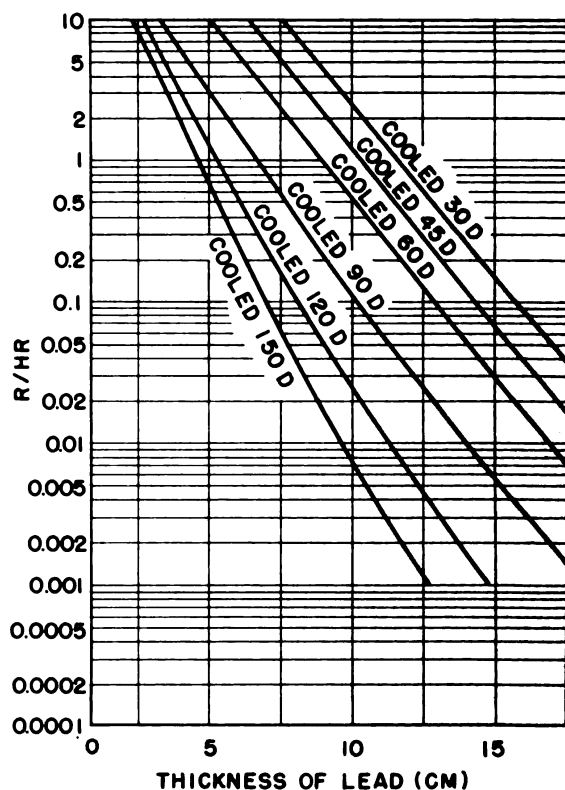


Fig. 2-54 Thickness of lead shielding for a 16-unit carrier.

rier are given in Figs. 2-54 and 2-55. Data are presented for iron and lead shields for various cooling times and emergent activities. Although a surface activity of 200 mr/hr is permissible for carriers, a maximum limit of 100 mr/hr has been selected to take into account the possibility of longer reactor use and short power surges above the rated 10-kw level. Thickness of shield and weight of carriers made from either iron or lead

are given in Fig. 2-56 for cooling times from 30 to 150 days.

OPERATING PROCEDURES

Routine. Routine operation is automatic, under the supervision of the operator. A key-operated switch is used by the operator to select the desired operation as OFF, 10 kw, or 100 kw. This sets up the correct range in the micromicroammeter (and hence the servo). It also sets the contacts (associated with the level safety channels) which monitor the power level and cause the rods to start in at 15 or 120 kw.

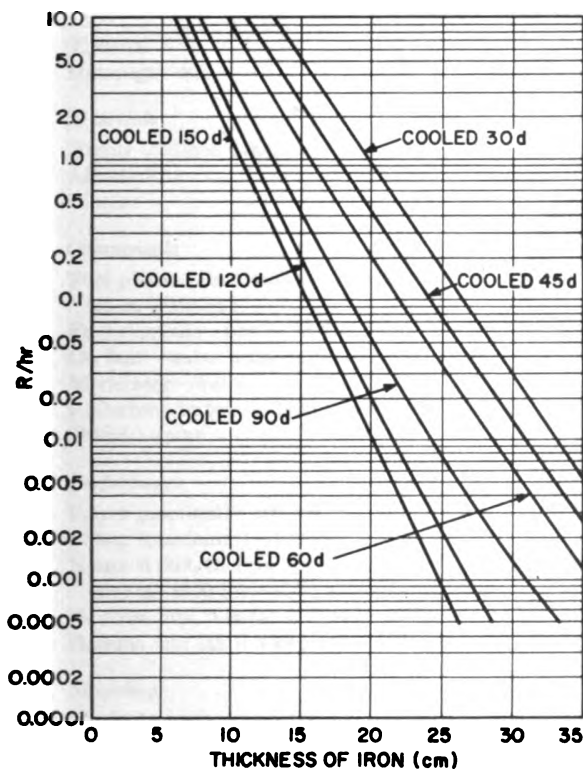


Fig. 2-55 Thickness of iron shielding for a 16-unit carrier.

Startup. A typical routine startup is as follows:

1. Turn key switch to 10 kw. Press OPERATOR SUPERVISION button. After the 5-sec warning has sounded, raise rods 2 in. (using manual controls) and periodically scram for routine check. A regular rotation should be established and the various scram signals should each be used in turn.

2. Press INSTRUMENT SUPERVISION button. The fission chambers are now inserted all the way.

Contacts in the counting-rate recorders will withdraw the chambers when the recorders go off scale. After the fission chambers are fully inserted, the counting rate is high enough so that the statistical variations in the derived period are not excessive. The motors then start withdrawing the rods, supervised by the period, which stops rod withdrawal when the period is shorter than 25 sec and inserts all rods if the period should become shorter than 7 sec.

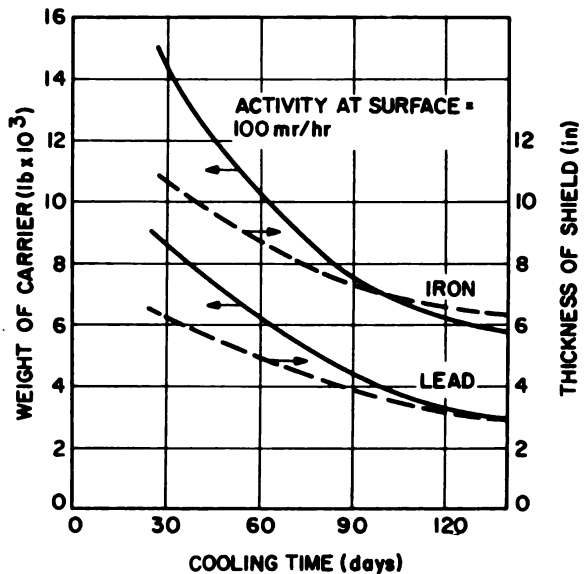


Fig. 2-56 Thickness of shield and weight of carriers.

3. After about 3 min, the subcritical rise and the transients associated with rod withdrawal will stop rod withdrawal an increasing number of times. Statistical variations in the observed period will result in an on-off pulsating withdrawal. The reactor will soon become critical and will rise with about a 20-sec period if sufficient reactivity is available. During this rise, the increase in counting rates will cause the fission chambers to be withdrawn, and the period channel period will supervise any further rod withdrawal needed.

4. At 10 kw, the servo will cause the shim-regulating rod to insert to level off the reactor power. If the shim rods are not fully withdrawn, they will continue to withdraw under the control of a timer, whose function is to ensure that insertion of the shim-regulating rod will not be overridden by simultaneous withdrawal of the two shim rods.

5. If the key switch is turned to 100 kw, the servo withdraws the shim-regulating rod, still supervised by the period, and brings the reactor to the new power. After running at 100 kw, the key switch may be returned to 10 kw, which will initiate a short reverse, and the power will stabilize at 10 kw.

6. The reactor should be shut down by initiating a scram, following the rotation prescribed above.

Manual. Whenever it seems desirable, the reactor may be run under the supervision of the operator; decisions are then made by the operator and actions are initiated and continued at his discretion, instrumentation permitting. This will be the normal method of operation during critical experiments and such other display experiments as seem best suited to this method of operation.

APPENDIX A

SPECIFICATIONS FOR BSR

1. *Type*

Thermal
Heterogeneous

2. *Materials*

Highly enriched uranium
Aluminum
Water

3. *Components*

Fuel plates—aluminum-uranium-aluminum (about $3 \times 24 \times 0.060$ in.)
Fuel elements—box of 18 fuel plates
Coolant—water between fuel plates
Moderator—water
Reflector—water (or BeO and water)
Shield—water

4. *Performance*

Power (nominal)—100 kw
Power (maximum)—1000 kw
Neutron flux, thermal (at 100 kw)— 10^{12} neutrons/cm²/sec;
(at 1000 kw)— 10^{13} neutrons/cm²/sec
Neutron flux, fast (at 100 kw)— 10^{12} neutrons/cm²/sec
Gamma flux (at 100 kw)— 8×10^6 r/hr

5. *Shielding*

Method—water of pool
Radiation level on bridge over reactor (at 100 kw)—
10 mr/hr

6. *Cooling*

Method—natural convection of water
Temperature rise of water through element—12°C at
100 kw; 37°C at 1000 kw

7. Demineralized water analysis

Concentration of all constituents in sample	Fresh water	3 months in pool	Units	Method
pH.....	6.3	6.6	pH	Meter
M.O. alkalinity as CaCO_3	2.0	6.0	ppm	Volumetric
Specific resistance at 18°C	1.17×10^5 *	1.12×10^5	ohms	Meter
Soap hardness as CaCO_3 detn.....	0.05	16.0	ppm	Volumetric
Soap hardness as CaCO_3 calc.....	0.05	0.84	ppm	Calculation from CaMg
Dissolved CO_2	1.0	1.0	ppm	Volumetric
Dissolved solids.....	4.3	7.7	ppm	Gravimetric
Nonvolatile solids.....	1.0	2.3	ppm	Gravimetric
SiO_2	0.05	0.57	ppm	Colorimetric
Fe.....	0.05	0.05	ppm	Colorimetric
Al.....	0.05	0.05	ppm	Colorimetric
Cu.....	0.05	0.08	ppm	Colorimetric
Ni.....	0.05	0.05	ppm	Colorimetric
Cr.....	0.05	1.0	ppm	Colorimetric
U.....				
Ca.....	0.05	0.34	ppm	Flame spectroscope
Mg.....	0.05	0.05	ppm	Colorimetric
Na.....	0.1	0.31	ppm	Flame spectroscope
SO_4	0.05	1.25	ppm	Colorimetric
Cl.....	1.0	0.42	pH	Meter
CO_3	1.2	3.6	ppm	Volumetric
HCO_3	2.4	7.32	ppm	Volumetric
NO_3	1.0	0.073	ppm	Colorimetric
PO_4	1.0	0.045	ppm	Colorimetric
F.....	0.05	1.0	ppm	Volumetric

* Reading at demineralizer was greater than 10^6 .

8. Composition of heavy concrete blocks used in pool

2210 lb barytes aggregate (BaSO_4) coarse

1050 lb barytes aggregate (BaSO_4) fine

1220 lb limonite ($2\text{Fe}_2\text{O}_3 \cdot 3\text{H}_2\text{O}$)

610 lb portland cement

46 gal water

(Quantities shown are for 1 yd^3 of mix.)

APPENDIX B

COMPONENT LIST: BSF OR SIMILAR FACILITY

<i>Item</i>	<i>Number required</i>	<i>Item</i>	<i>Number required</i>
Core:			
Grid plate.....	1	ing period amplifier, sigma amplifier, and magnet amplifier).....	1
Standard fuel elements.....	Set (20-35)	Strip chart recorders.....	4
Control-rod fuel elements.....	3-4	1024 scaler.....	1
Reflector elements, BeO.....	Set (5-26)	Log <i>N</i> amplifier.....	1
Reactor bridge and supporting structure:		Compensated ion chamber (differential).....	3
Bridge.....	1	Gamma chamber.....	1
Support for grid plate and chambers.....	1	Chamber power supplies.....	4
Bridge lock.....	1	Log count-rate meter.....	1
Hand crank for moving bridge.	1	Micromicroammeter external shunts.....	1
Instrument bridge and supporting structure:		Galvanometer.....	1
Bridge.....	1	Electrometer (gamma).....	1
Hand crank for moving bridge.	1	Fission chamber and drive mechanism.....	1
Carriage for supporting and positioning instruments.....	1	A-1 amplifier and preamplifier (fission chamber).....	1
Safety and control rods and actuating mechanisms:		Relays.....	
Regulating rod.....	1	Safety chambers.....	2
Safety rods.....	2-3	Monitron.....	1
Magnets.....	5-6 (including spares)	Manually operated scram switches.....	6
Magnet guides.....	3-4	Health-physics instruments:	
Magnet amplifier.....	3	Counter-scaler with recorder and immersion probe for monitoring the build-up of induced activity in the water	1
Rod drive mechanism.....	3-4	Mobile counter-scaler with beta-gamma probe mounting for verification of freedom from contamination of clothing, etc.....	1
Selsyns.....	6-8 (3-4 transmitter types) (3-4 receiver types)	Beta-gamma portable survey meter for quick determination of beta- and gamma-radiation levels.....	2
Servo.....	1	Air monitor for alpha particles	1
Magnetic amplifier for servo...	1	Monitrons.....	3
Indicator lights.....		Measuring instruments	
Key switch.....	1		
Safety and control instruments and circuits:			
Panel or console.....	1		
Sigma amplifier and preamplifier.....	2		
Composite amplifier (contain-			

<i>Item</i>	<i>Range</i>	<i>Item</i>	<i>Range</i>
Thermal-neutron detectors:		Zinc sulfide scintillation dosimeter...	
1/4-in. fission chamber (U^{235}).....	10^6 to 10^8 nv		10^6 to 10^8 nv
3-in. fission chamber (U^{235}).....	5×10^6 to 10^8 nv	Gamma-ray spectrometer:	
8-in. BF_3 counter.....	5×10^6 to 10^8 nv	Unique-response multiple-crystal	
12 1/2-in. BF_3 counter.....	10^6 to 10^{-2} nv	scintillation spectrometer using	
(Two of the 12 1/2-in. size are used		pair process and Compton effect	
in parallel to provide this sensitiv-		(developed at NEPA and ORNL)	
ity)		Fast-neutron spectrometer:	
Indium and gold foils.....	10^7 to 10^8 nv	U^{238} fission chamber (detects neutrons	
Gamma-ray dosimeters:		above 1 Mev)	
50-cm ³ ion chamber.....	4000 to 10^{-3} r/hr	Proton-recoil spectrometer using alu-	
900-cm ³ ion chamber.....		minum absorption of protons (de-	
Anthracene dosimeter (developed at		veloped at NEPA and ORNL)....	
ORNL)		10^7 to 2×10^4 nv	
Fast-neutron dosimeter:		Nuclear plate camera (developed at	
Hurst dosimeter (developed at		Los Alamos and ORNL).....	
ORNL).....	10^6 to 10^8 nv		

APPENDIX C

DRAWINGS

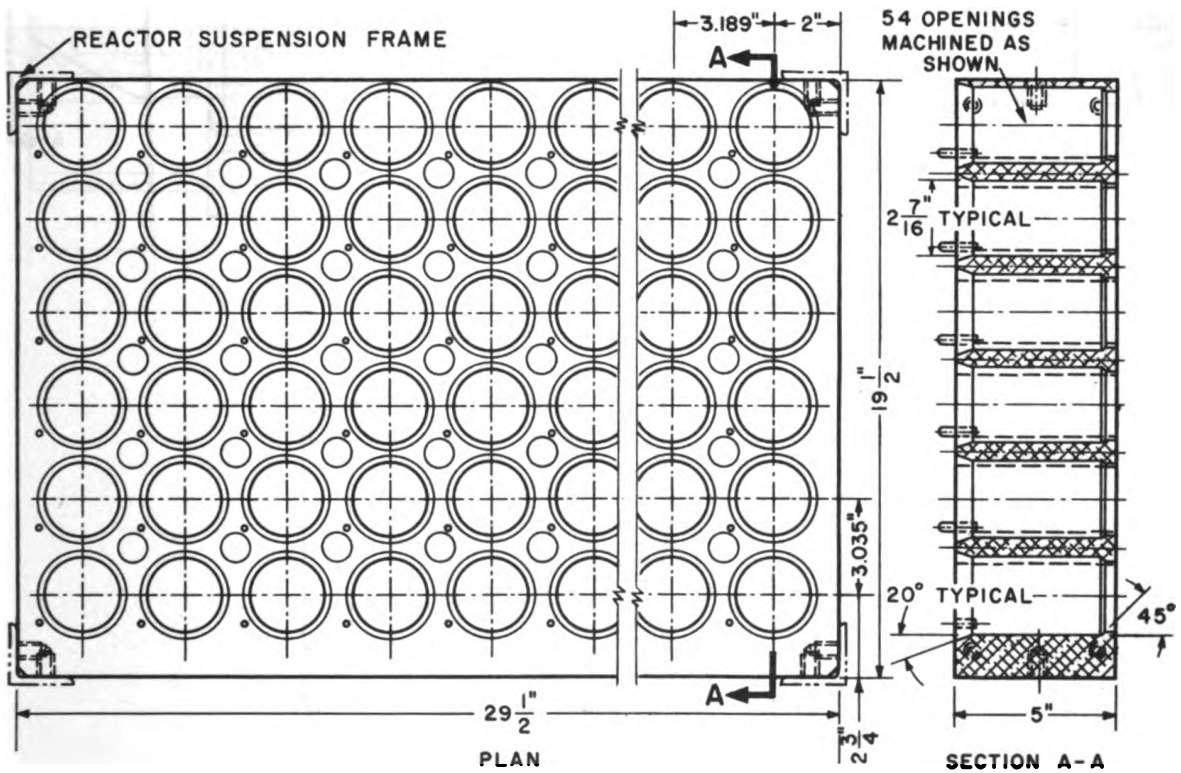


Fig. 2-57 Grid plate.

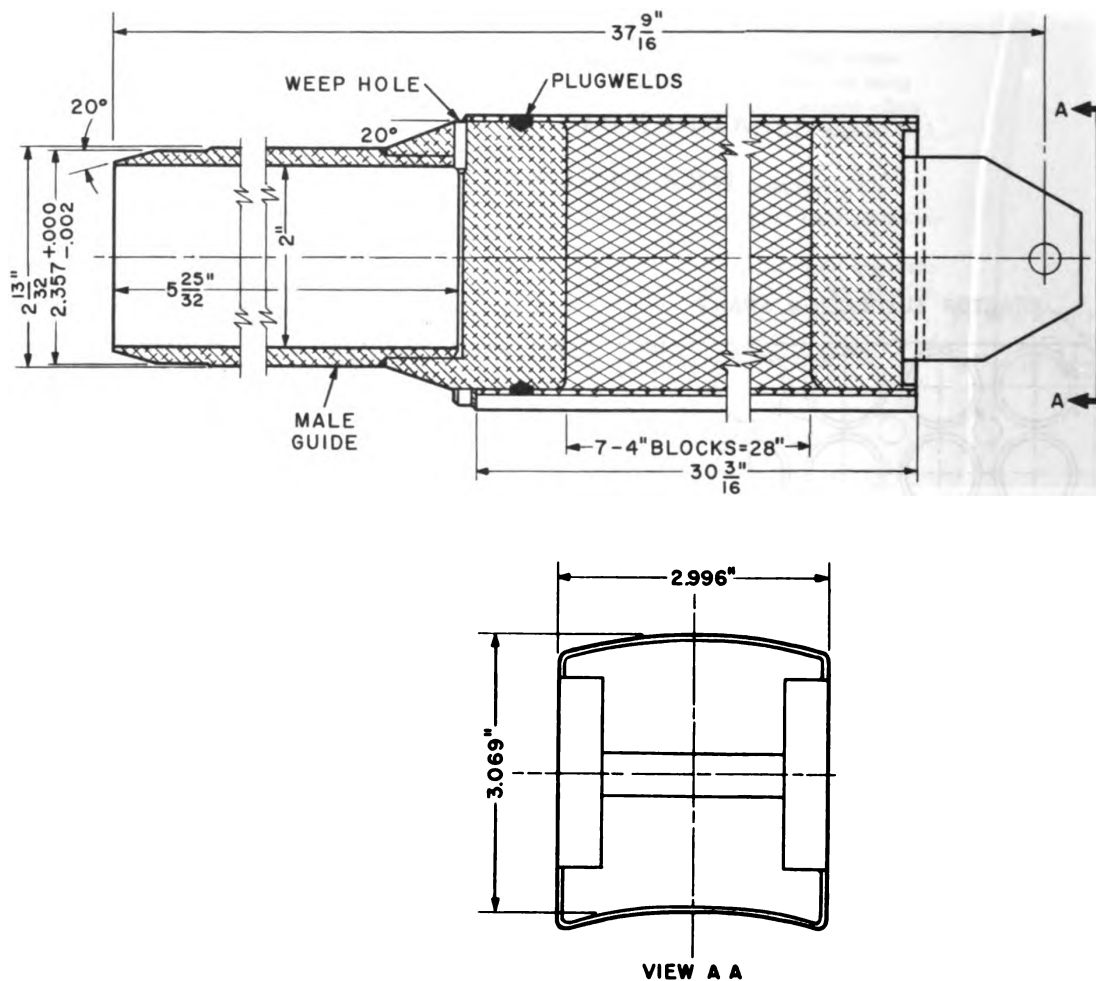
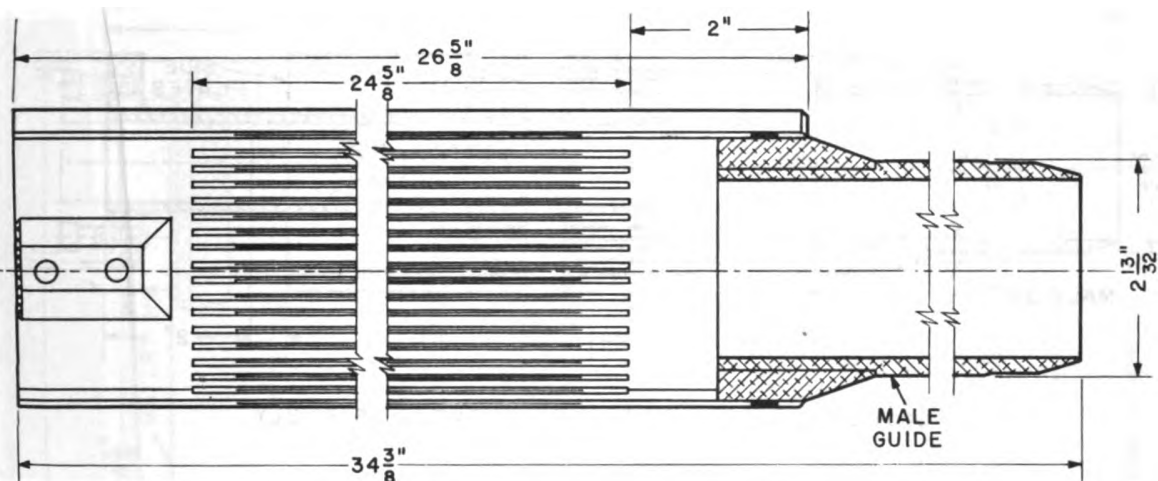


Fig. 2-58 Small reflector element.



SECTION A-A OF FUEL ELEMENT ASSEMBLY

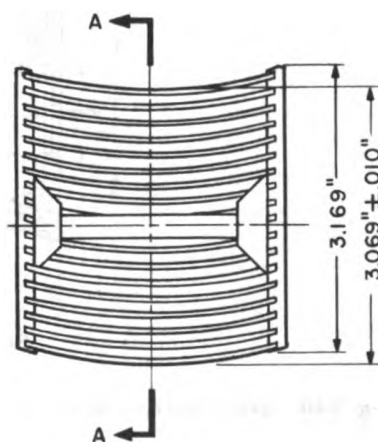
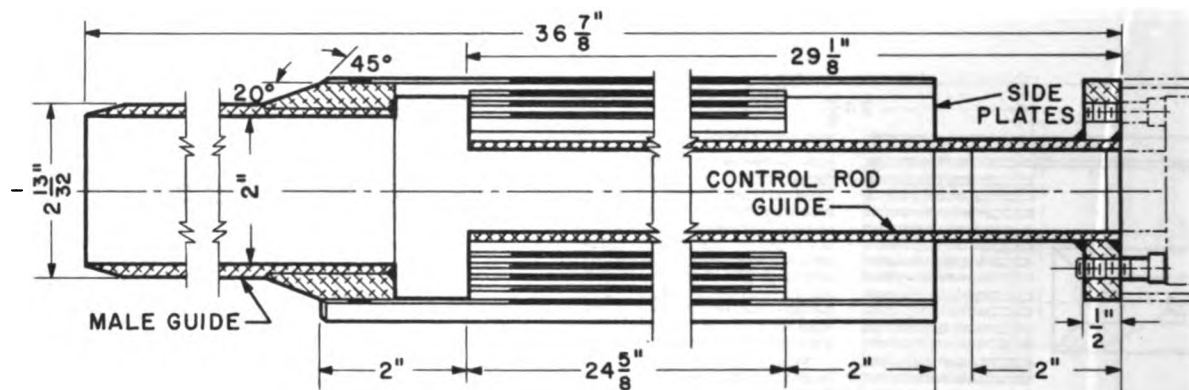


Fig. 2-59 Standard fuel element.



SECTION A-A OF SPECIAL FUEL ELEMENT ASSEMBLY

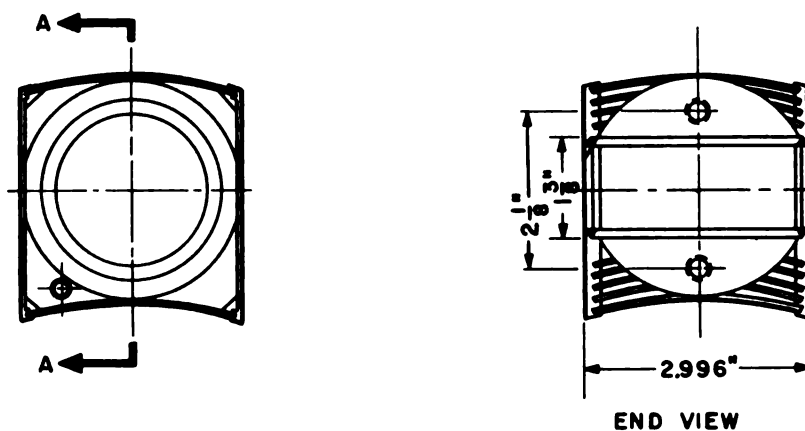


Fig. 2-60 Special fuel element for control rod.

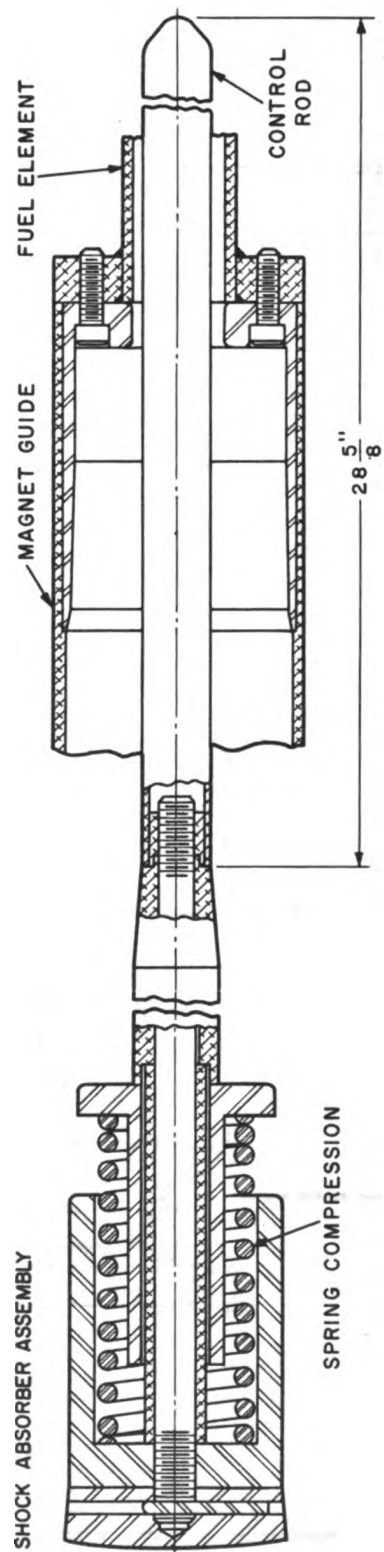
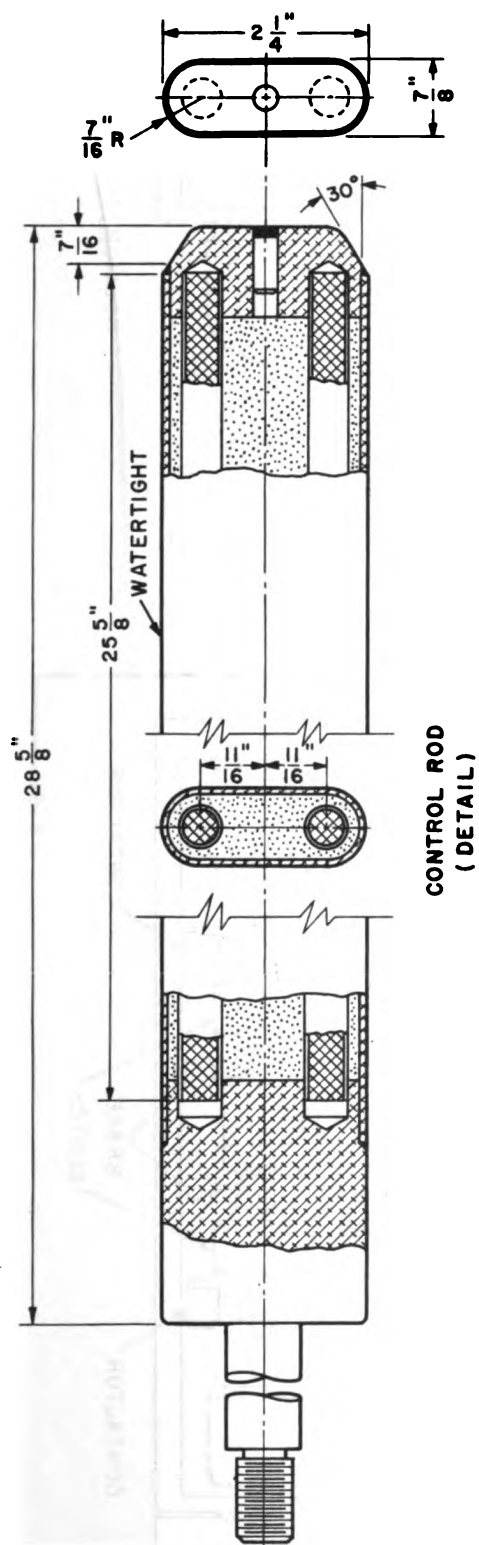


Fig. 2-61 Control rod, magnet, and shock absorber.

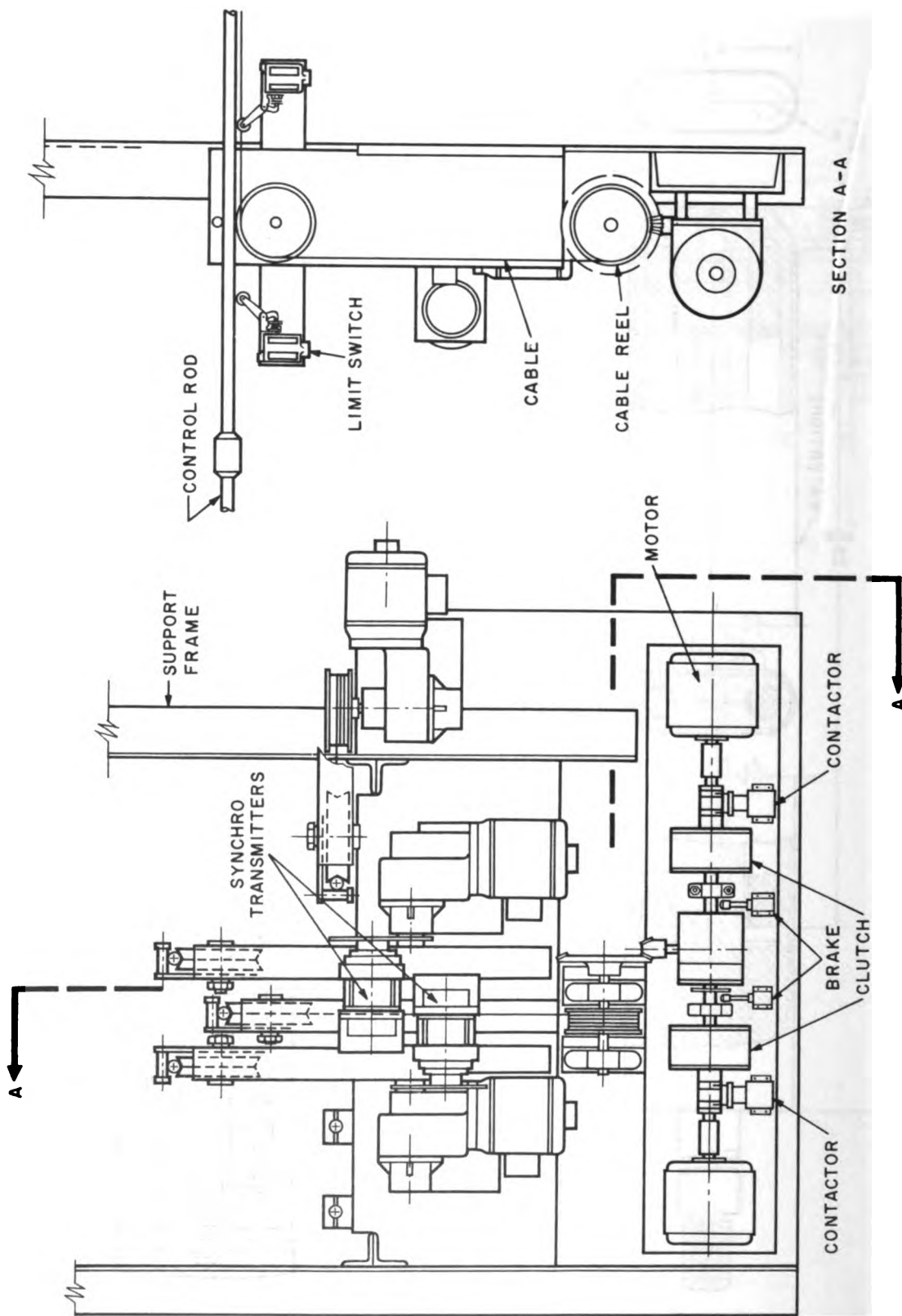


Fig. 2-62 Motor drive for reactor controls.

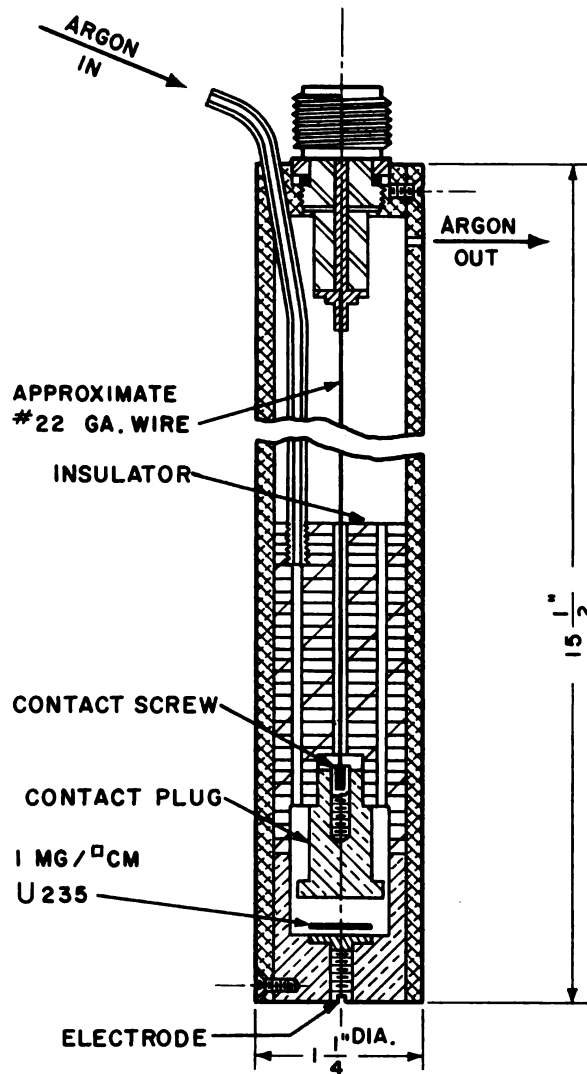


Fig. 2-63 Fission chamber, $\frac{1}{2}$ -in. diameter.

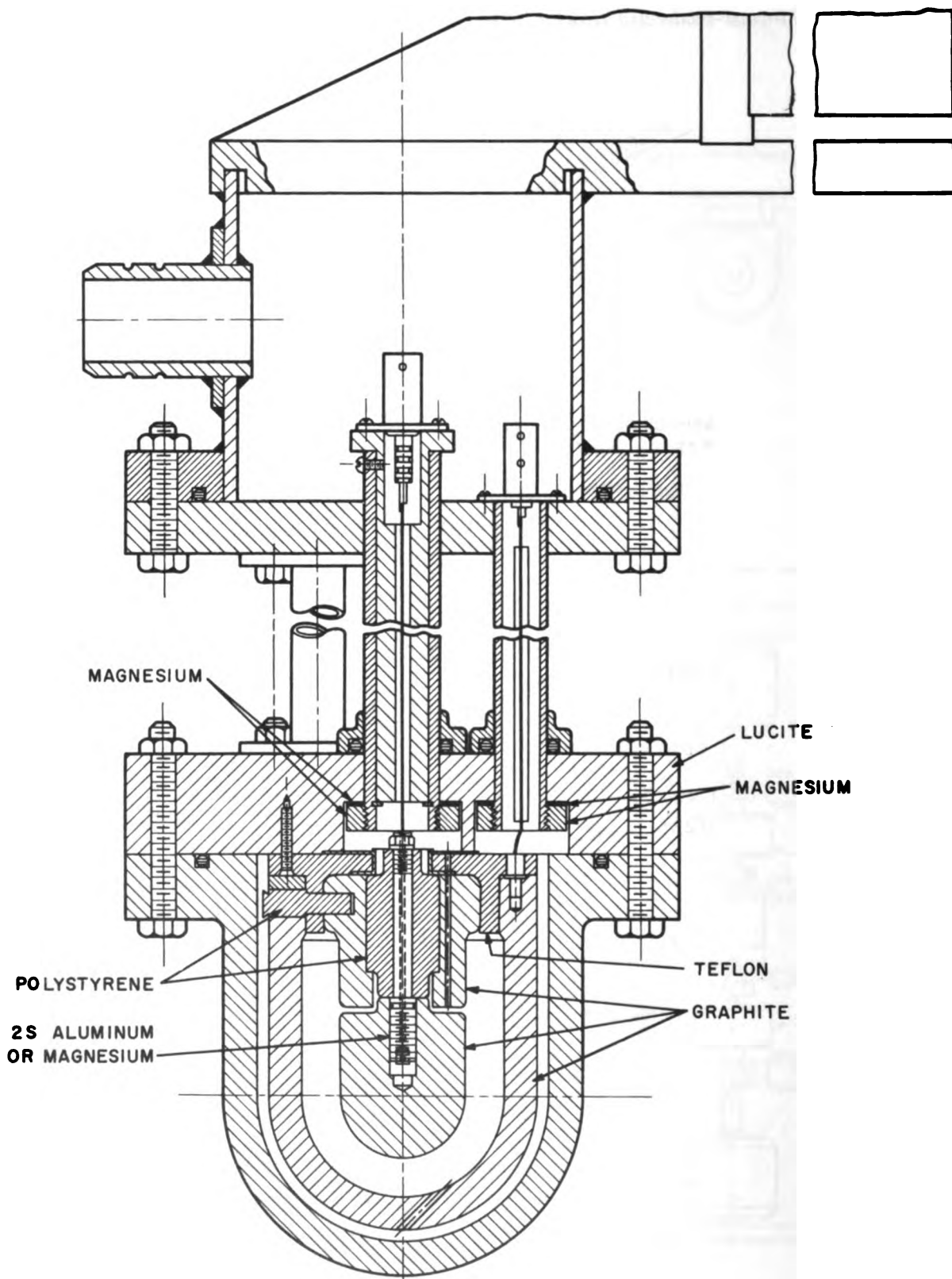


Fig. 2-64 50-cm³ ion chamber.

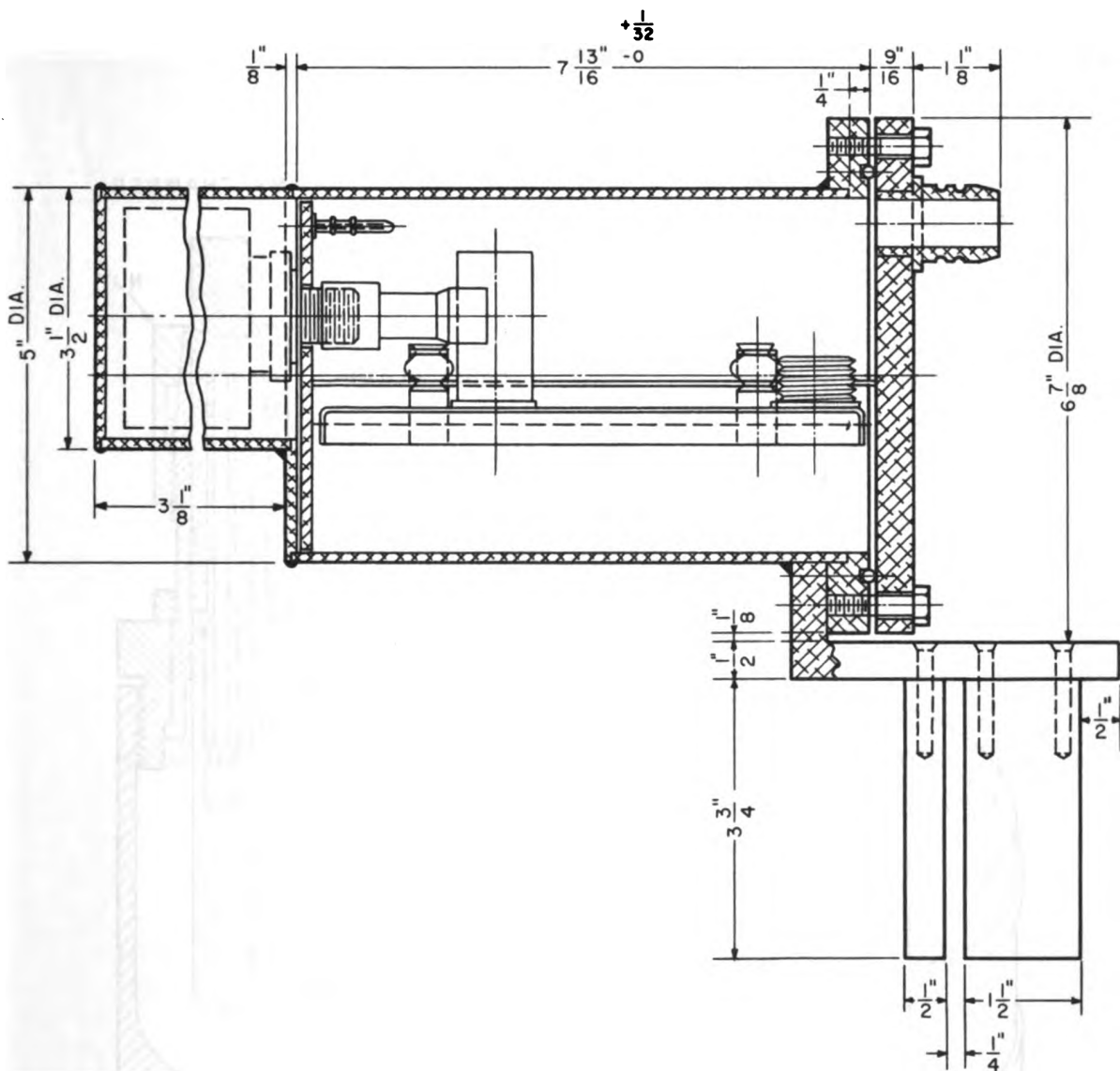


Fig. 2-65 Watertight housings.

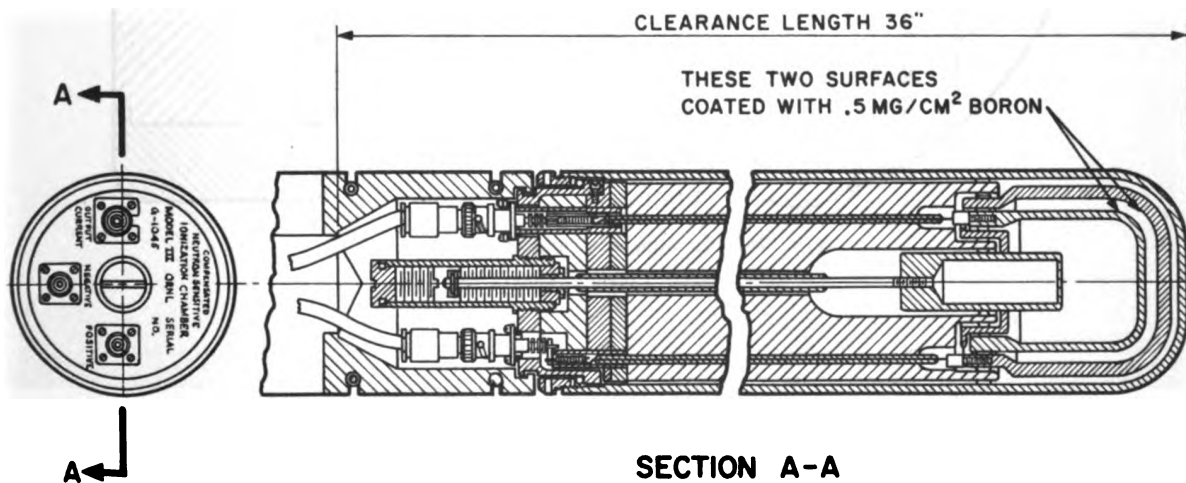


Fig. 2-66 4-in. compensated-ion chamber.

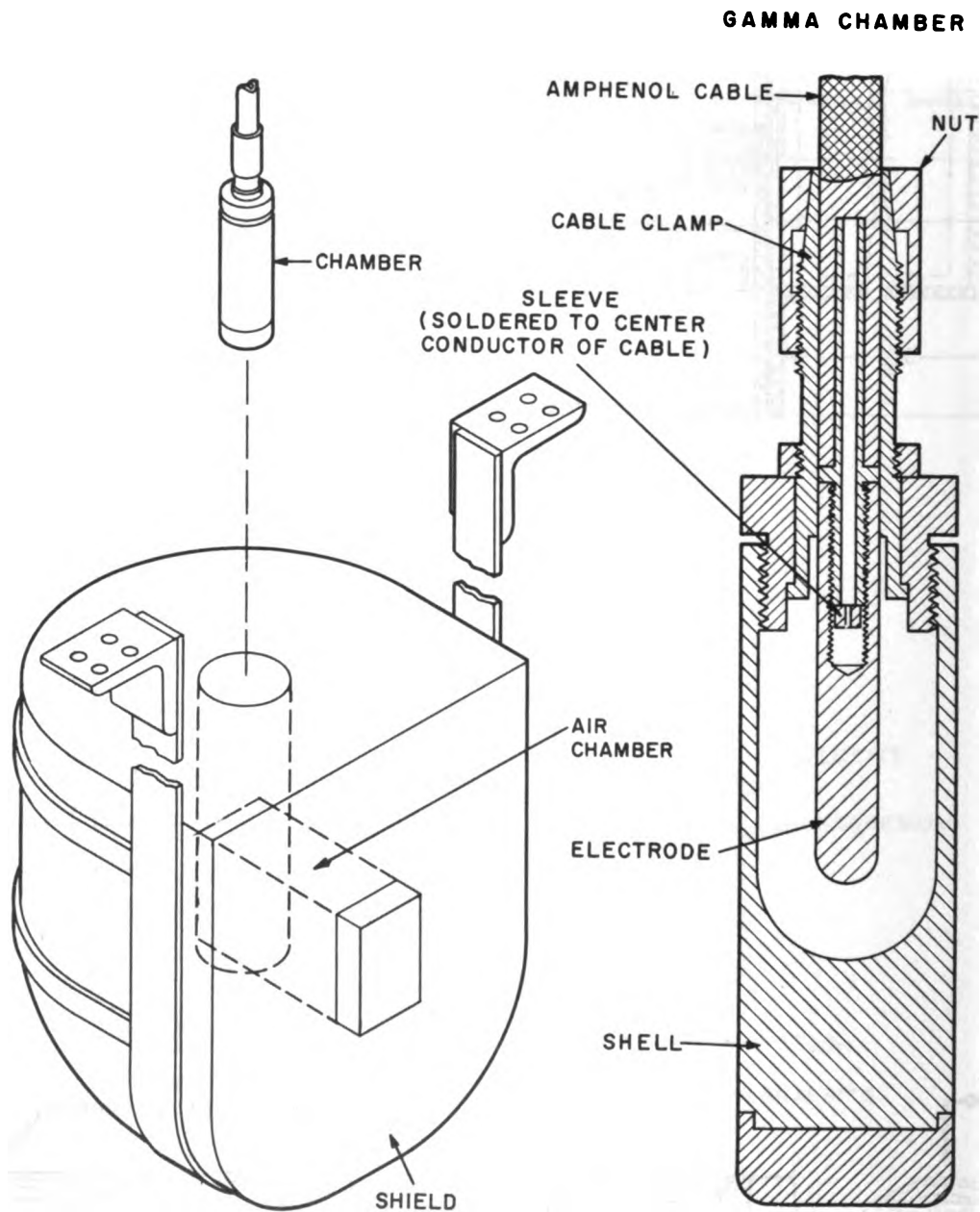


Fig. 2-67 Gamma ion chamber and shield.

CHAPTER 3

Light-water-moderated Reactor

TYPE III *Heterogeneous—Enriched Fuel*

COMPILED BY NATIONAL REACTOR TEST STATION
PHILLIPS PETROLEUM COMPANY

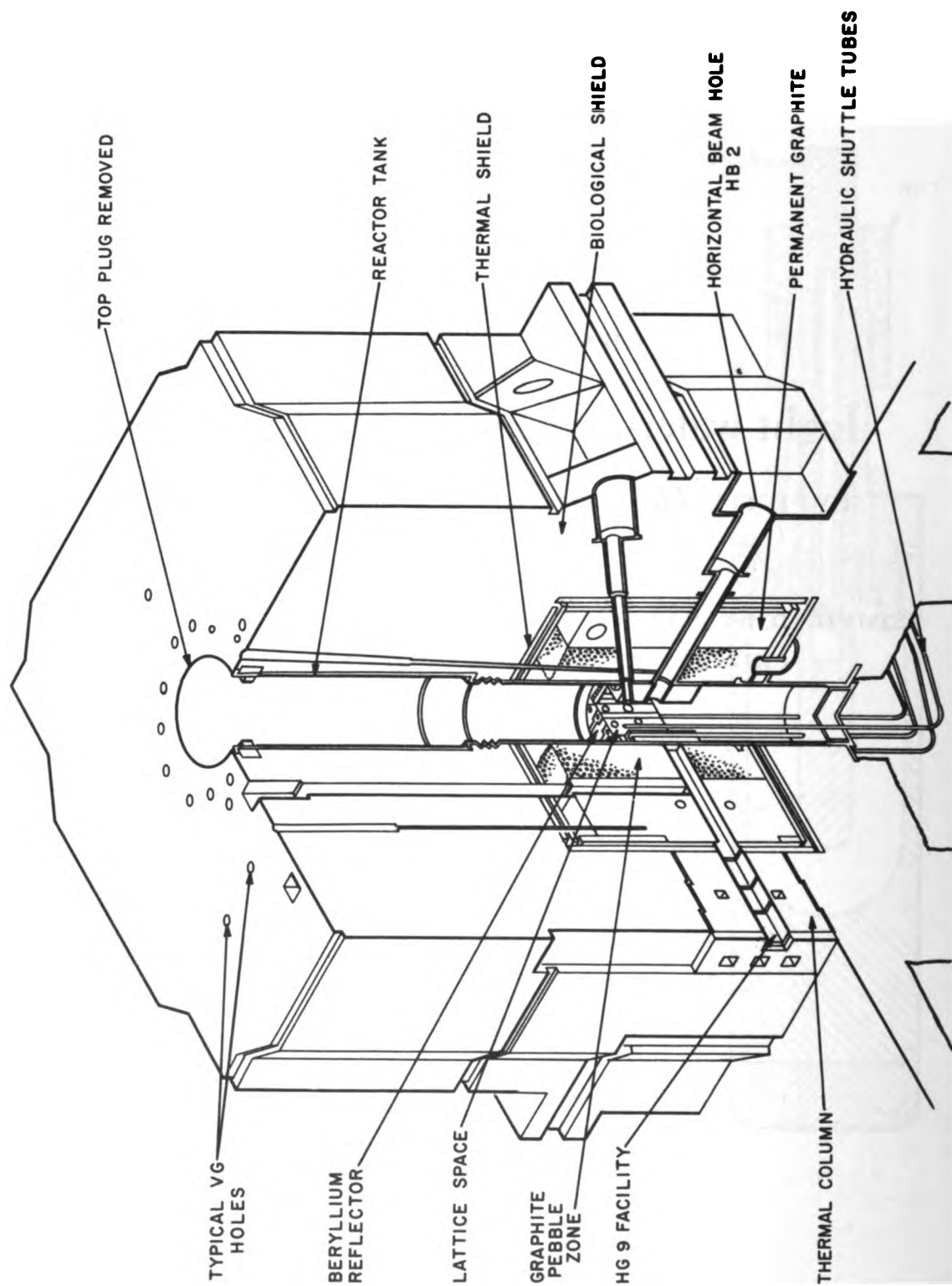


Fig. 3-1 Quarter-section cutaway view of materials-testing reactor.

CHAPTER 3

Light-water-moderated Reactor

TYPE III *Heterogeneous—Enriched Fuel*

PART 1 DESIGN

3-1 Brief General Description. The materials-testing reactor (MTR), as its name implies, is designed to provide facilities for testing materials in high-intensity radiation fields. It has the highest neutron flux of any known reactor now in existence and has operated very satisfactorily since March, 1952, without significant loss of time due to shutdowns except those which constituted a regular part of the operations program.

The MTR is a thermal-neutron reactor using enriched uranium as fuel, ordinary water as both moderator and coolant, and beryllium as reflector. It is designed to operate at 30,000 kw. Because of its high specific power, average neutron fluxes of 2×10^{14} thermal neutrons/cm²/sec and 1×10^{14} fast neutrons/cm²/sec are available.

Peak thermal-neutron fluxes of 5×10^{14} neutrons/cm²/sec exist in certain positions in the reflector.

The enriched-uranium fuel is contained in an "active core" which is located inside a "lattice" region having the nominal dimensions of 40 by 70 by 60 cm. high. It is surrounded by a beryllium reflector 3 ft high and roughly 5 ft in diameter. Both the lattice and the reflector are enclosed in an aluminum tank 5 ft in diameter, which is extended top and bottom to form a "well" 30 ft deep (see Figs. 3-2 to 3-5). This well is closed top and bottom with heavy lead-filled steel plugs. The top one is removable when the

reactor is not running so that the fuel in the lattice and the beryllium in the reflector can be loaded or unloaded. It also serves to hold the driving mechanisms for the control rods.

The loading and unloading of reactor fuel and other material inside the tank is accomplished through 20 ft of water by using hand grappling tools to pick up the radioactive components. These are lifted just above the beryllium reflector and moved to a position directly over a hole through the reflector, which is filled with a beryllium plug during operation of the reactor. Beneath this hole is a water-filled discharge tube leading to a receiving tube in a deep canal under the reactor. At the time of unloading, the plug is removed. The discharge tube which leads to the canal connects with the receiving tube through a valve and seal which act as locks to prevent excessive loss of water from the reactor. This receiver can be rotated hydraulically to a horizontal position to discharge the radioactive component under 15 ft of water in the canal. New fuel units and nonradioactive parts are loaded by lowering them directly through the water from the top of the reactor structure.

In order to provide additional space having thermal-neutron fluxes above 10^{13} neutrons/cm²/sec, a secondary reflector of graphite is placed outside the reactor tank. It is divided into two zones: a pebble zone that is adjacent to the reac-

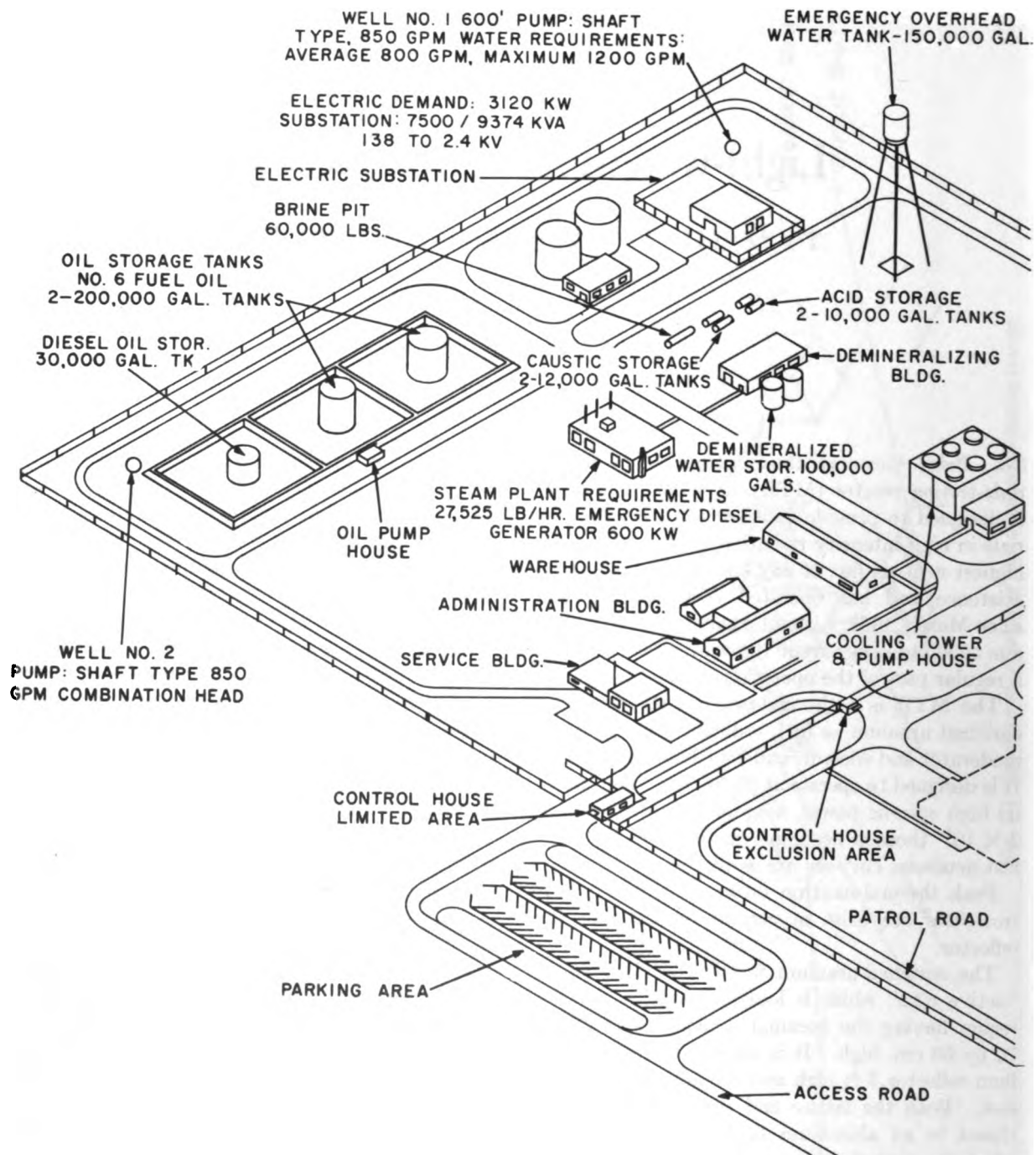
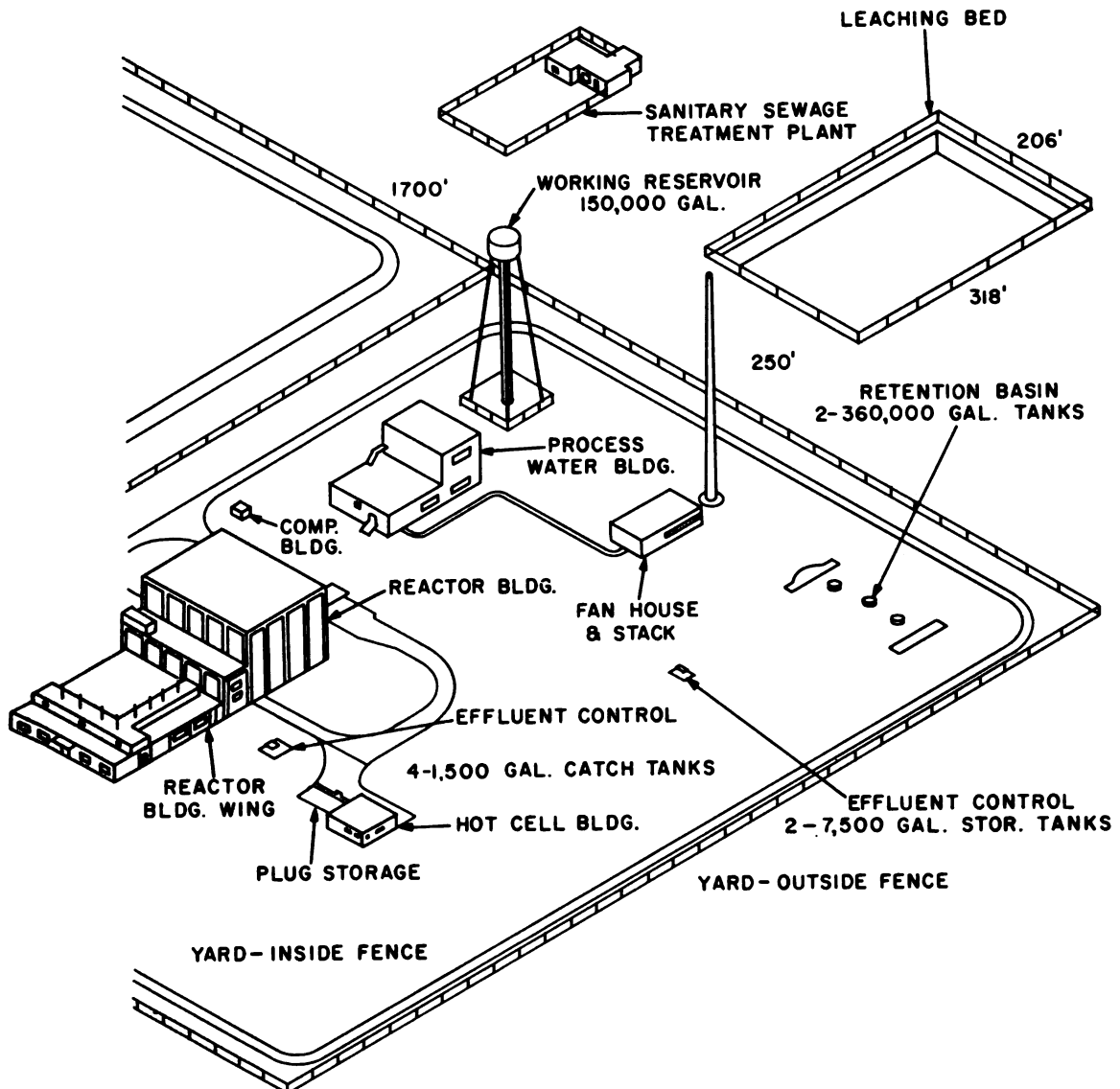


Fig. 3-2 Isometric view of the

WATER TREATMENT DOMESTIC:
SOFTENING FIRE AND IRRIGATION:
RAW PROCESS: DEMINERALIZE TO 3PPM



materials-testing reactor site.

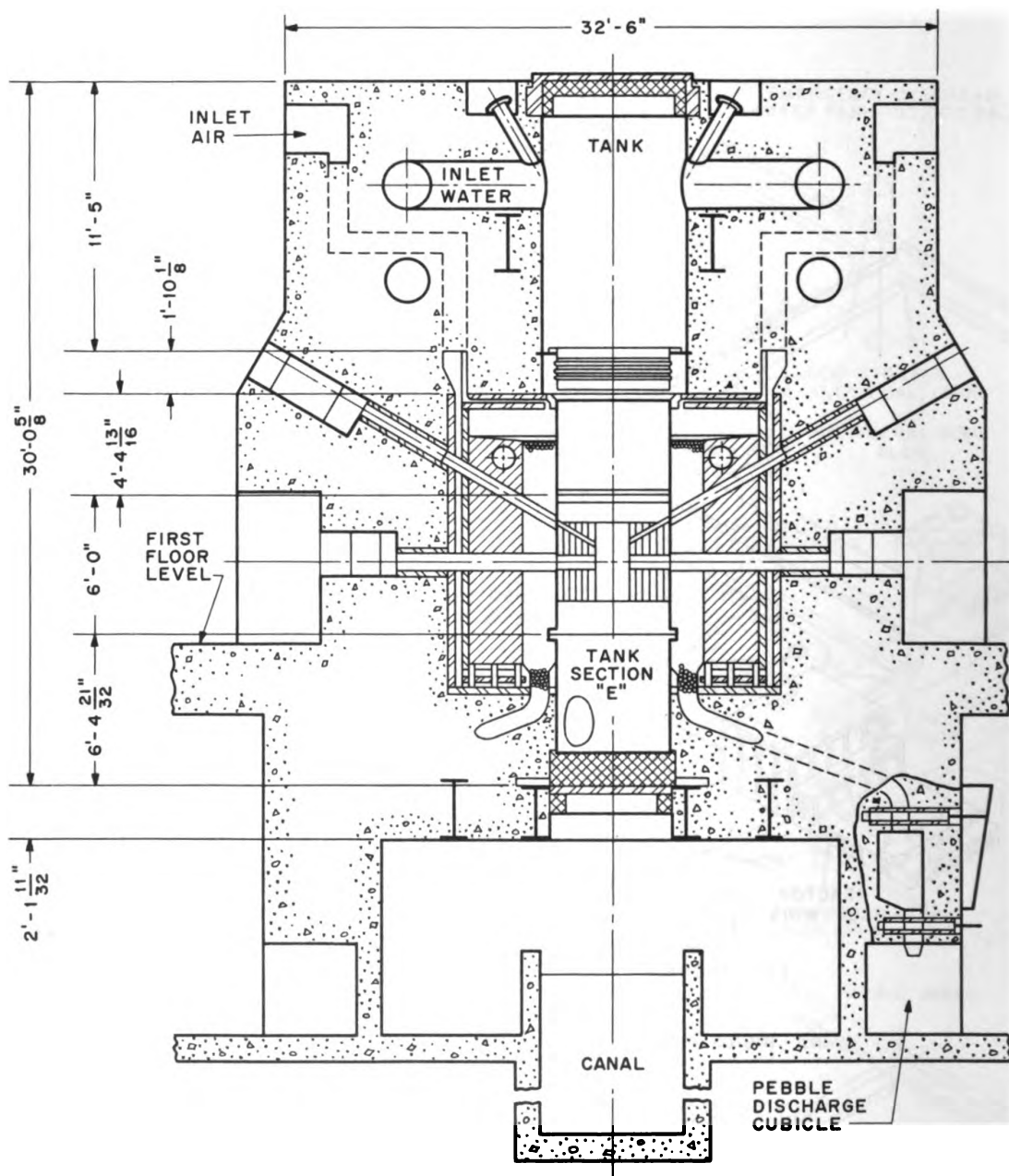


Fig. 3-3 North-south vertical section of the materials-testing reactor.

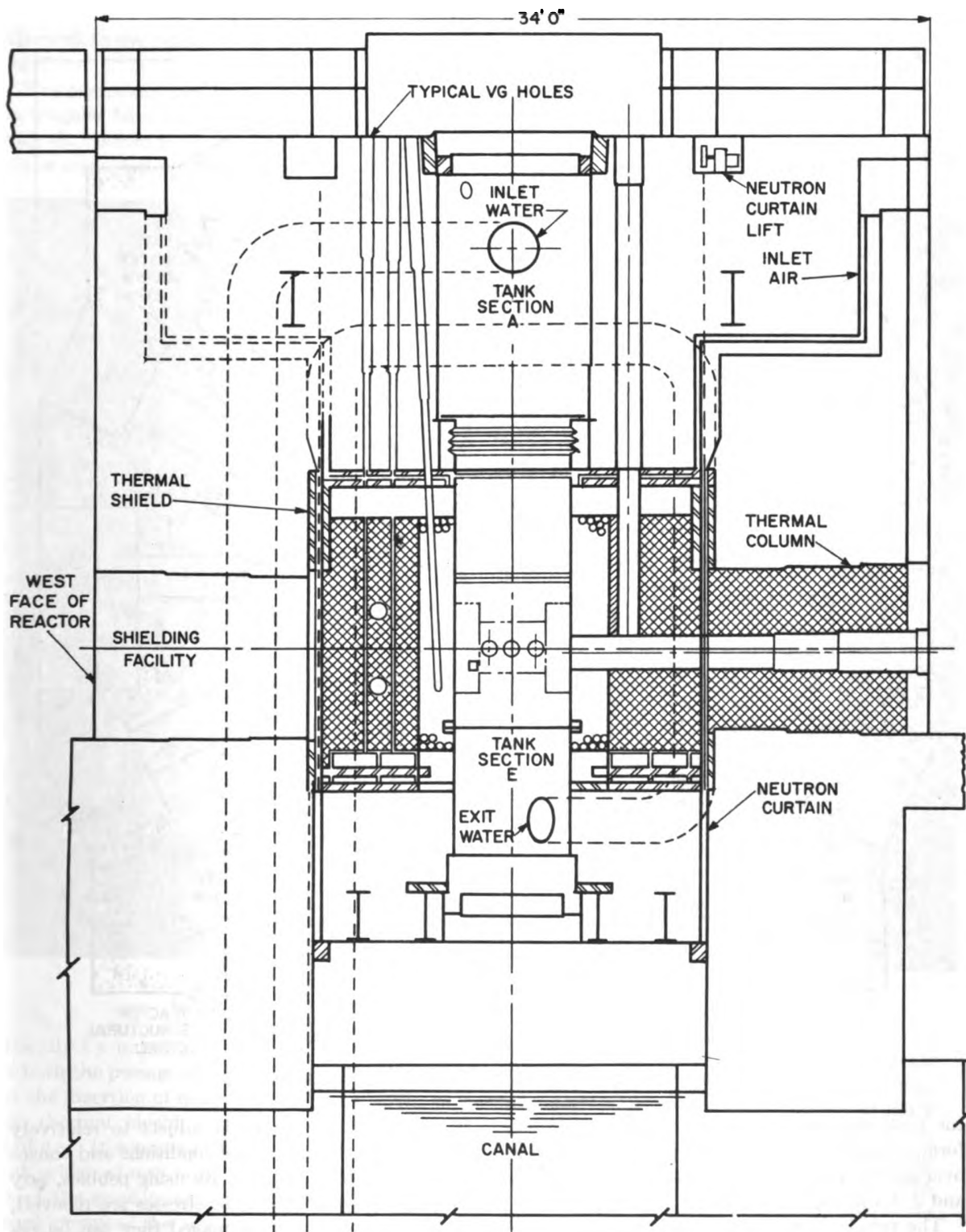


Fig. 3-4 East-west vertical section of the materials-testing reactor.

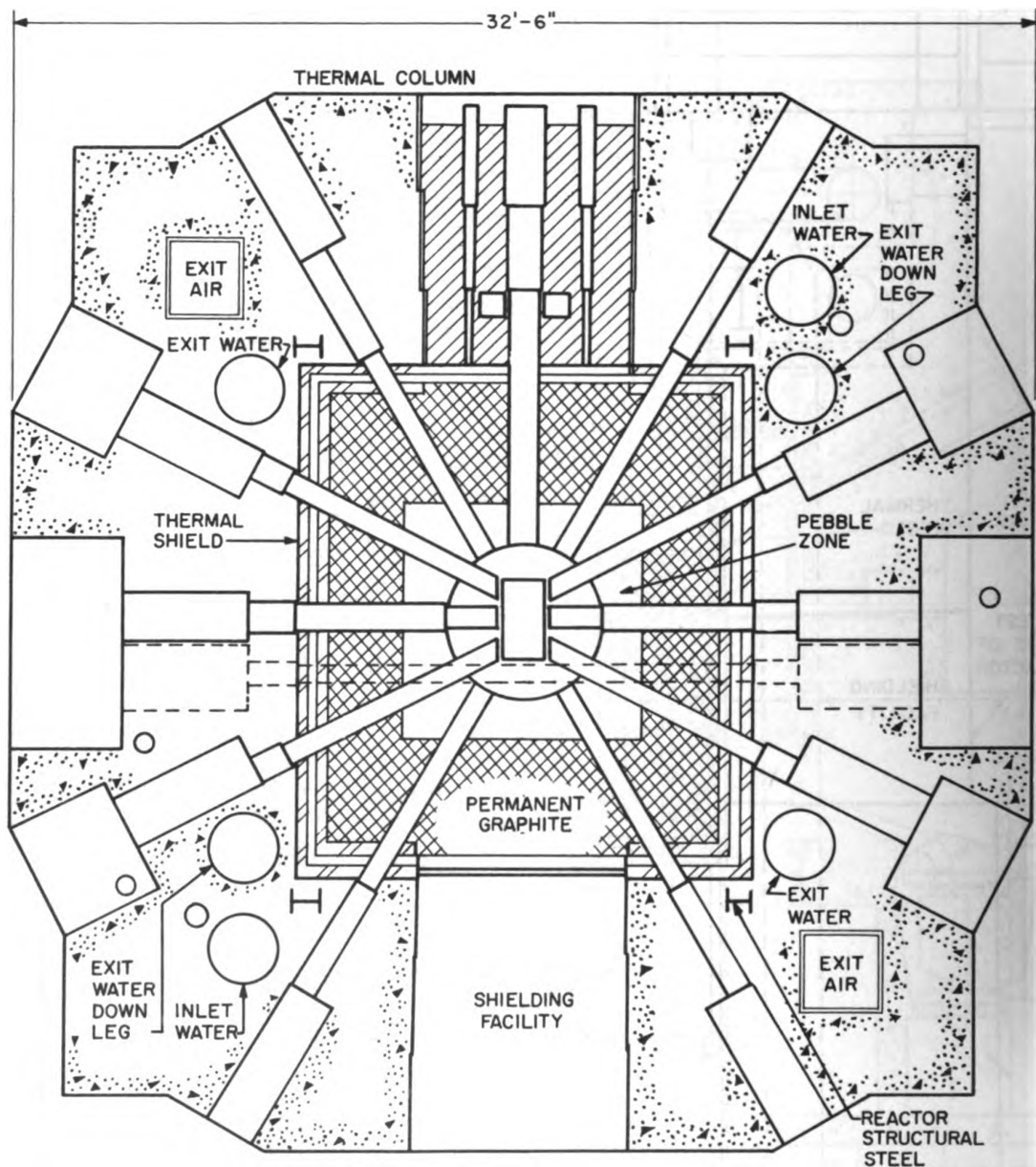


Fig. 3-5 Horizontal section of the materials-testing reactor.

tor tank and extends 40 cm from the tank, to form a square, and a permanent zone that is 12 ft over all north to south, 14 ft over all east to west, and 9 ft 4 in. high.

The pebble zone is a volume 7 ft 4 in. square and 9 ft high. It surrounds the reactor tank and is filled by about 700,000 graphite pebbles 1 in.

in diameter. This zone is subject to relatively high heat and temperature conditions and consequent possible oxidation. By using pebbles, any expansions, contractions, or stresses are relieved, and if the pebbles are damaged they can be removed through discharge chutes and replaced. In addition, the problem of cooling the graphite

adjacent to the reactor tank is reduced considerably.

The permanent graphite is composed largely of rectangular bars stacked to a height of 9 ft 4 in. with the bottom located 4 ft below the center line of the active lattice. This structure is penetrated

Finally, a concrete shield about 9 ft thick is provided to reduce the level of radiation to a value small enough so that a low instrument background is obtained near the reactor structure. The outer dimensions of the structure form roughly a cube 32 ft on a side.

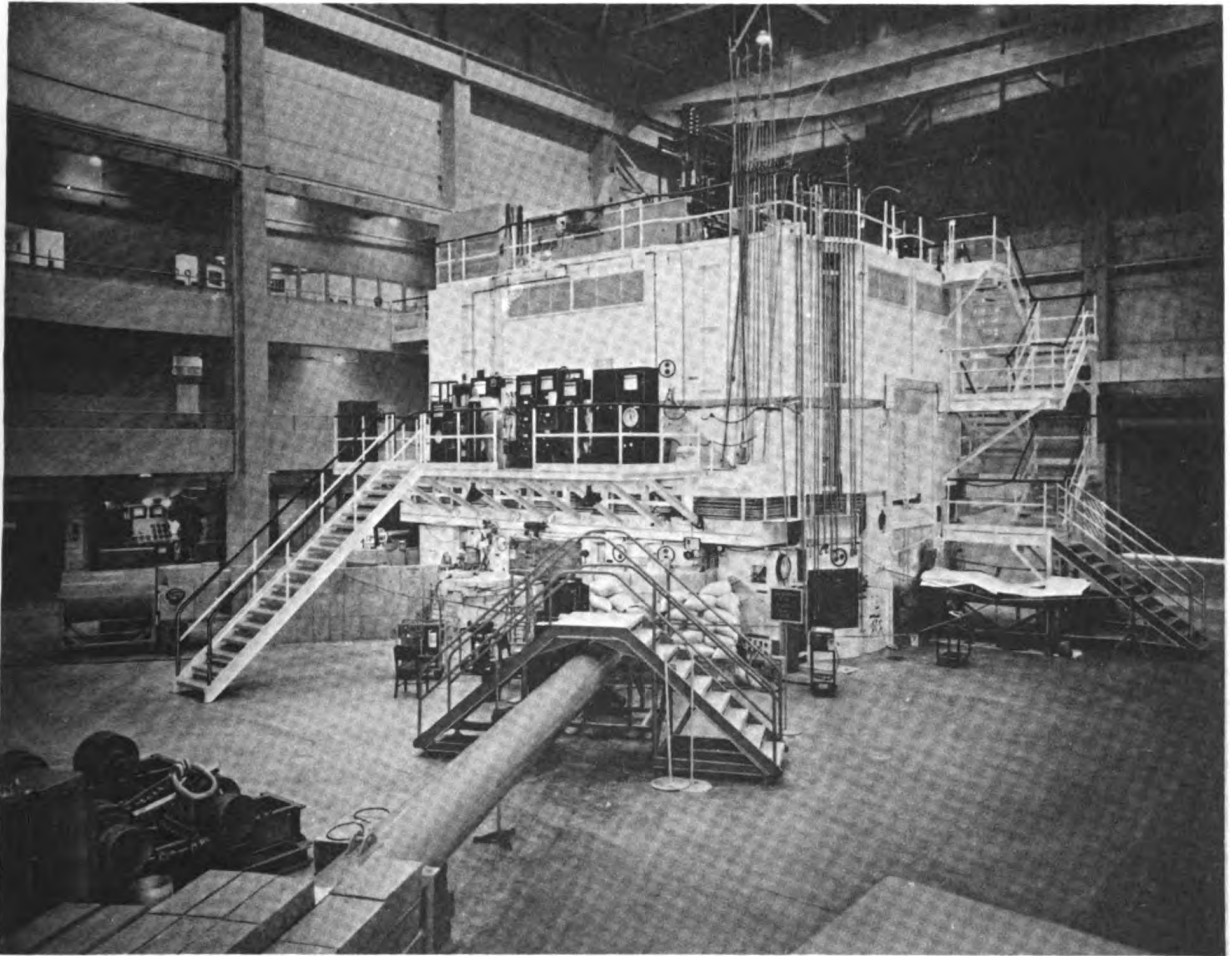


Fig. 3-6 View of the south and west faces of the materials-testing reactor.

vertically by numerous holes which serve to permit both the passage of air for cooling the graphite and the insertion of materials to be irradiated.

A thermal shield completely surrounds the graphite. It consists of two layers of steel 4-in.-thick which absorb most of the neutrons leaking out of the graphite and in which most of the residual energy of the radiation from the reactor is given up. They are separated by 4 in. in order to provide an air passage for cooling.

The reactor itself and the beryllium reflector are cooled by water flowing at a rate of 20,000 gpm. This water enters near the top of the well at 100°F and leaves near the bottom at 111°F. It is fed by gravity from a head tank, 170 ft high, through the reactor tank to a vacuum-spray evaporator system for cooling. The water is then elevated to the head tank by centrifugal pumps with a capacity of 10,000 gpm. Purge water flows to a retention basin to permit radioactive decay and then passes

to the surrounding terrain. A secondary circulating stream of water from a cooling tower is used in the condensers of the spray evaporators.

The graphite reflector and thermal shield generate about 500 kw of heat by absorption of neutrons and gamma rays. They are cooled by air flowing

ceived for the express purpose of facilitating the conception and design of future reactors. It is important that a facility exist which is capable of furnishing quickly much physical, chemical, biological, and engineering information at thermal- and fast-neutron fluxes greater than 10^{14} and

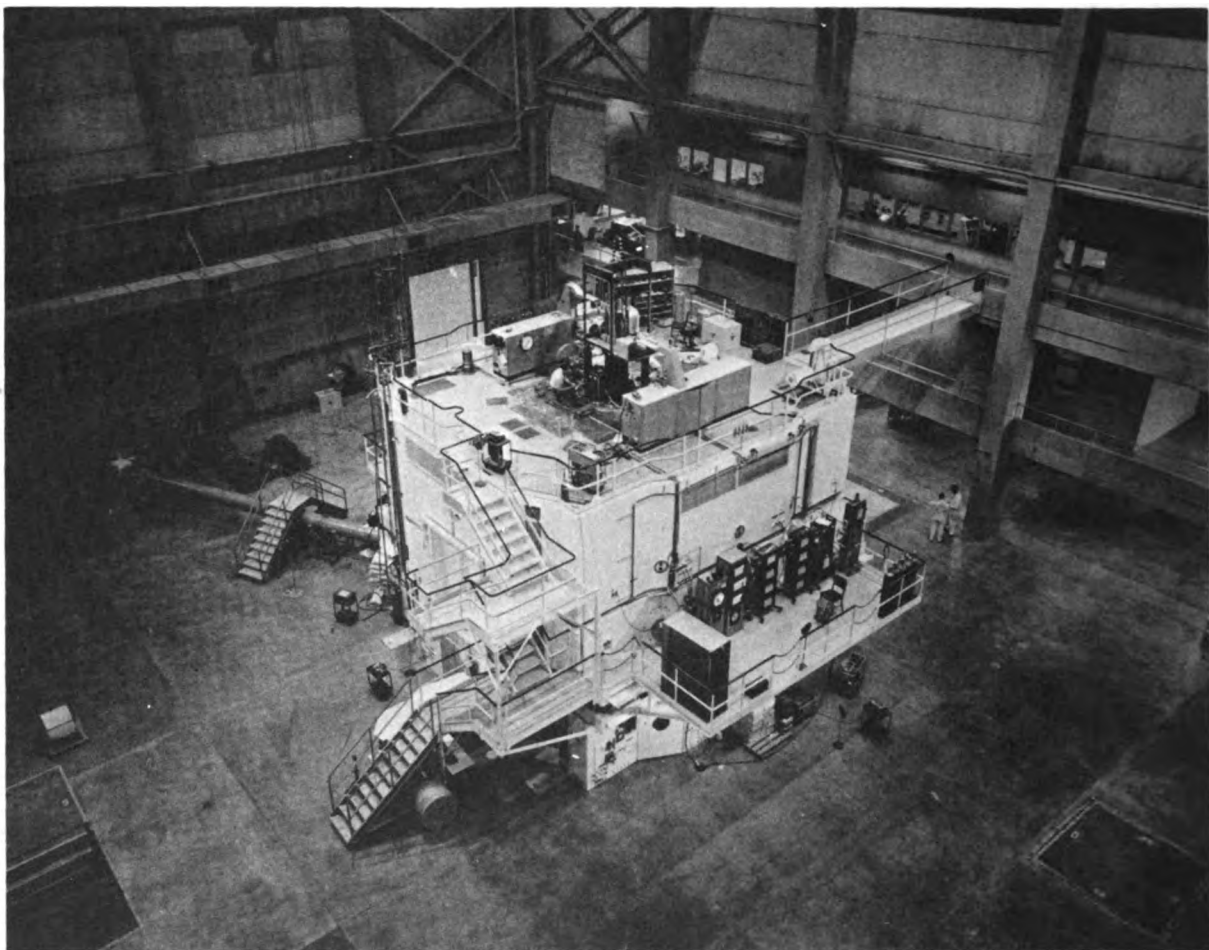


Fig. 3-7 View of the north and east faces of the materials-testing reactor.

at a rate of 1700 lb/min. Large blowers in a fan house create a suction in the reactor structure so that the air is taken from the Reactor Building and discharged to the atmosphere through a stack 250 ft high.

Exterior views of the reactor are shown in Figs. 3-6 and 3-7.

3-2 Need and Uses of the Materials-testing Reactor. The MTR is the result of an evolutionary process in the design of a machine con-

high gamma-ray intensities, on the basis of which future reactors can be designed within controllable and permissible limits of uncertainty in operating characteristics.

A reactor which is to be most effective as a radiation source should be designed to maximize both fast- and thermal-neutron fluxes. To obtain these goals a reactor must have a high specific power per gram of fissionable material and per unit of volume.

As reactor technology enters these fields of

higher fluxes, many problems arise which require experimental answers. For example, the effect of high fluxes, particularly of fast neutrons, on the properties of structural materials is still not well known. Some structural materials, such as graphite, are known to suffer ill effects in a relatively short time. Metals which have desirable structural properties as a result of cold working or of heat-treatment may conceivably undergo adverse effects under fast-neutron bombardment. Also, the cross section of the fission products may determine to a large extent the breeding gain or the operating time of a power reactor. It is possible that second-order reactions may be important and that nuclei of short life and, at present, unknown properties may thus be disproportionately important in a high-flux reactor as compared with their importance at lower fluxes.

Aside from these highly urgent and immediately practical problems, there are the many fundamental experimental studies in high-flux regions which can be carried out in the MTR. The high density of epithermal neutrons in the MTR permits the extension of the practical range of the crystal spectrometer to higher neutron energies. The properties of fissionable isotopes in the lower-resonance energy regions may thus be investigated and their nuclear constants more accurately determined. Again, reactions of higher order (the capture of two or more neutrons in succession by the same nucleus) can be made to produce significant quantities of rare isotopes, thereby permitting the study of their nuclear constants—heretofore impossible in many cases.

Another use of the MTR is for the production of radioisotopes in amounts and of specific activities heretofore not readily available. For example, the MTR can produce phosphorus compounds of high specific activity which are usable directly, without purification or chemical separation, by clinical institutions. Strong Co^{60} sources are also easily produced.

The MTR provides thermal- and fast-neutron flux levels for component testing and materials irradiation which are high enough to extrapolate to future reactors under consideration. It also provides several experimental holes designed to permit the irradiation of reactor components, such as fuel assemblies, under simulated operating conditions of temperature, pressure, and par-

ticular coolant. To produce these high-neutron and heat fluxes, however, means a sacrifice in volume for experimental facilities and causes high fuel burn-up and operational difficulties.

3-3 A Brief Design History. Originally, a 50-kw homogeneous uranium heavy-water reactor was proposed whose main use would be to produce easily extractable fission products. Work proceeded for a short while until it was decided that the power should be raised sufficiently to demonstrate the feasibility of a thermal-energy breeder. The flux in such a reactor was calculated to be about 2×10^4 neutrons/cm²/sec.

Certain problems arose in connection with the conception of control and technology of a homogeneous reactor, and it was then decided to proceed with another type of reactor which would produce similar flux conditions.

A heavy-water-moderated and heavy-water-cooled enriched heterogeneous reactor was first designed. Later, on the basis of theoretical calculations, a design proposal was submitted for a high-flux thermal reactor of heterogeneous core which would be light-water-cooled and heavy-water-moderated and -reflected. The use of light water as a coolant was believed preferable to the use of heavy water because light water increased the operational flexibility of the machine.

On critical examination of the foregoing design, it was found that an important gain in fast flux could be obtained by using light water for both coolant and moderator. This change would result in a smaller reactor, and because fast (virgin) flux is proportional to power per unit volume, a 30,000-kw reactor of the proposed type could produce a fast flux of approximately 1×10^{14} neutrons/cm²/sec and a thermal flux (2×10^{14} neutrons/cm²/sec) about equal to that of the previous design. In addition, a further volume decrease was gained with light-water moderation.

Actually, in the previous design with heavy-water moderation and reflection, the reflector merely provided space for experiments that could be accomplished in other ways. It was therefore decided to eliminate heavy water altogether and to make the reflector of beryllium. This decision was made possible by the rapidly improving metallurgical techniques of handling and fabri-

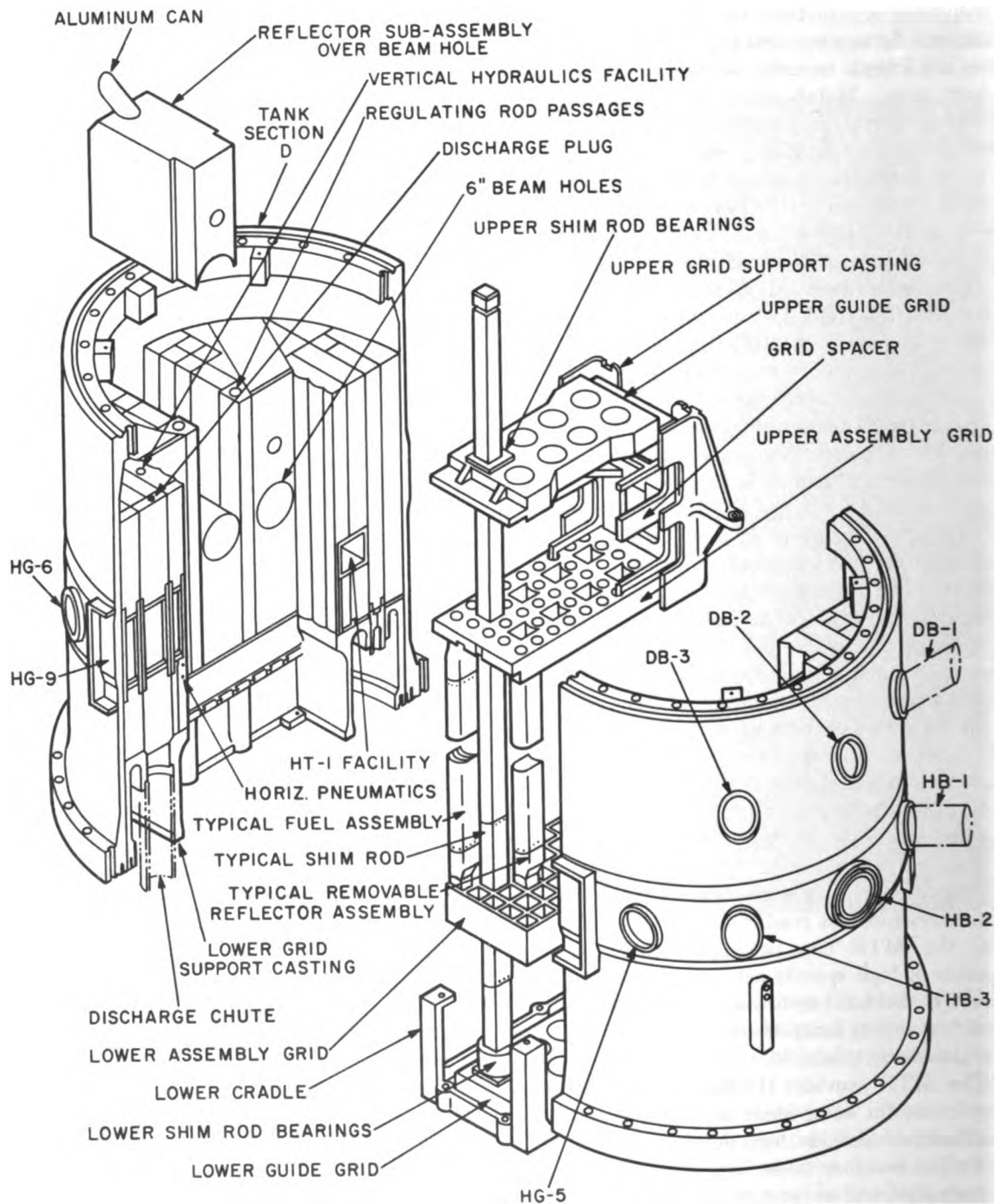


Fig. 3-8 Pictorial cross section of the reactor tank.

cating this metal on a production basis ranked in tons rather than pounds.

Thus the design for the experimental high-flux reactor evolved from an initially homogeneous fluid reactor to an all-metallic rigid machine, aside from the cooling fluid, which was ordinary water.

Summary of costs for completed plant. Total costs for the materials testing reactor were \$17,160,000.

The following is a general breakdown of the total cost:

Yard areas inside and outside perimeter fence (includes parking lot; all underground piping, services, utilities; sidewalks, roads, grading, etc.)	\$ 2,337,200
Reactor and Reactor Building	7,333,800
Reactor Wing Building (laboratories, offices, change rooms, etc.)	1,920,800
Process-water Building, Working Reservoir, Cooling Tower and Pump House, and Demineralizer Building	2,840,200
Emergency Overhead Water Tank, Reservoir and Pump House, Water Supply, Wells, etc.	439,700
Fan House and Stack	442,500
Retention Basin, Effluent Control	531,900
Plug Storage, Services Building, Canteen, Office Building, Warehouse Building	289,600
Steam Plant, Substation, Primary Substation, Feeder Power Line, Sewage Disposal, Compressor Building	1,011,900
Limited-area and Exclusion-area Control Houses	12,400
Total	\$17,160,000

ENGINEERING DESCRIPTION OF THE REACTOR

The Reactor

The general design of the reactor is shown in Figs. 3-2 to 3-8, which are an isometric view, a north-south vertical section, an east-west vertical section, a horizontal section, two photographs, and a pictorial cross section, respectively. Over-all dimensions are given in Figs. 3-3 and 3-4, but it should be noted that in all construction drawings the horizontal center line of the reactor (i.e., the center line of the horizontal experimental holes of Figs. 3-3 and 3-4) is arbitrarily taken as elevation 100 ft. On this basis the first floor is at elevation 96 ft 6 in. and the top of the reactor is at elevation 120 ft 6 in.

3-4 Reactor Tank System. The general tank system, approximately 30 ft tall, is shown in Figs. 3-3 and 3-4, while the details of the construction, gasketing, etc., are shown in Fig. 3-9. The tanks provide the means of funneling water through the active lattice and beryllium and provide supports for the active lattice, the reflector, and the control-rod bearings. They are bolted together and with aluminum gaskets make a watertight system.

Tank sections A and E. Both of these tank sections are made of stainless steel and are permanently embedded in the concrete of the biological shield. However, they were initially supported independently of the concrete on a structural-steel frame. The bottom of this frame is shown under tank section *E* in Fig. 3-4, and the vertical members are shown just outside the thermal shield in Fig. 3-5. Both tank sections were welded to this frame before the concrete was poured.

Tank section *E* (Fig. 3-3) is required in the structure both to act as a water exit under the active lattice and to provide space for the control-rod extensions. For easier construction and smooth water flow, it is made the same diameter as tank section *D*.

Tank section *A* (above tank section *E*) (Fig. 3-3) serves as (1) an entrance for the cooling-water stream and (2) a container for the radiation shield (water) used during shutdown operations. During shutdown approximately 17 ft of water is required as a cover for the active lattice. Tank section *A* is built with water inlets and overflow lines near the top to fulfill this requirement. It will also be noted that tank section *A* is large enough in diameter to permit mounting the lighter tank sections *B*, *C*, and *D* after tank sections *A* and *E* have been finally set, and in case of damage, the section permits replacement of these tanks.

The proper operation of the control rods depends upon the alignment of the tank sections, and it is essential that the top flange of tank section *E* be set accurately horizontal and that tank section *A* and all the beam holes in the concrete be aligned accurately to this flange.

Tank section B. Tank section *B* is a bellows-type expansion joint, made up of two convolutions of thin stainless steel welded to stainless-steel flanges for attachment to tank sections *A* and *C*. This structure permits free thermal ex-

pansion and contraction of the tank sections without applying excessive stress to the structural setup. Removal of this section permits filling of the pebble zone.

Tank sections C and D. These tank sections are essentially one in function but are separate for ease in manufacturing. The split between *C* and *D* is dimensioned so that the support castings for the active lattice can be mounted conveniently. These tank sections are hollow aluminum cylinders of

wall. These six thimbles are extruded aluminum tubes of 6 in. ID and $\frac{5}{16}$ in. wall thickness, sealed at the end adjacent to the lattice by an aluminum plate $\frac{7}{8}$ in. thick set parallel to the lattice face. They are welded into the tank wall in position to receive the ends of the HB liners. As described in Sec. 3-7, the beryllium reflector is machined to fit around these thimbles with sufficient clearance for proper flow of cooling water (see also Sec. 3-15).

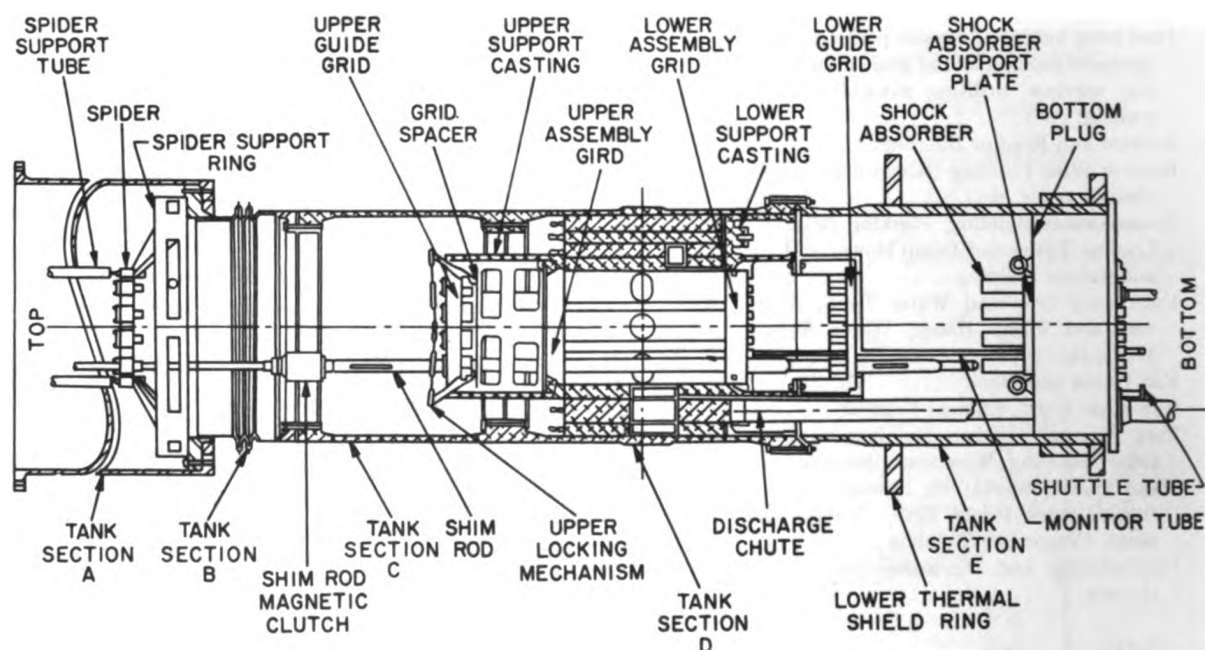


Fig. 3-9 Section of reactor-tank assembly.

1 in. wall thickness and $54\frac{1}{4}$ in. ID. The diameter was determined by the requirements of the active lattice and the beryllium reflector. The wall thickness was designed to give the strength required to sustain the water pressure and the weight of the support castings, but it is made as thin as possible, consistent with these requirements, so that the neutron absorption will not be too great. In general the tanks have a wall 1 in. thick, except for a 1-ft band around the center of *D* tank where the wall is $1\frac{1}{4}$ in. thick.

Tank section *D* is pierced by several holes to provide experimental facilities close to the active lattice. Most of these, for the six HB (horizontal beam-hole) thimbles, are at the center line of the lattice and pierce the thick section of the tank

In addition to the thimble holes, tank section *D* is pierced by the aluminum bar containing the HR-1 and HR-2 (pneumatic shuttle¹) holes and by the HT-1 (horizontal through hole) facility. The aluminum bar containing the shuttle holes is shown in Figs. 3-8 and 3-16. It reaches to the bottom of the reflector only to simplify the beryllium machining and stacking. The shuttle holes themselves pass just under the extension of HG-9 (horizontal hole in the graphite) and within 1 in. of the east face of the active lattice.

The HT-1 facility through the tank is a square aluminum tube with approximately $4\frac{3}{4}$ in. inside dimensions and walls $\frac{1}{2}$ in. thick. It runs

¹ HR stands for "horizontal rabbit." "Rabbit" is used colloquially for "shuttle," and not in the biological sense.

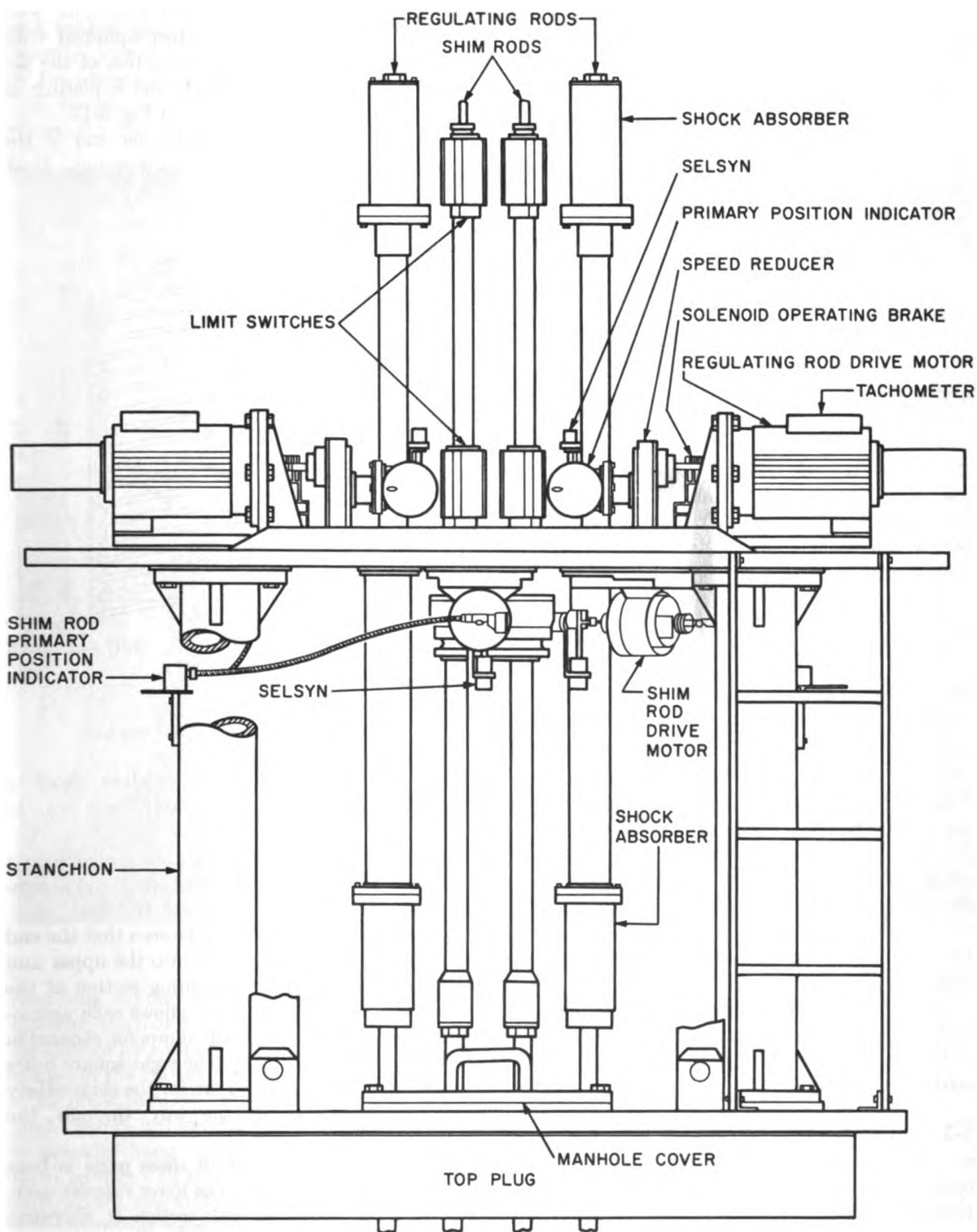


Fig. 3-10 Top-plug assembly.
165

just under the beam holes close to the west face of the active lattice, as shown in Fig. 3-8. Adjacent to the lattice, the inner wall is extended down to the bottom of the reflector to facilitate the packing of the beryllium. Both the bar containing HR-1 and HR-2 and the HT-1 tube are welded into tank section *D*, keeping it a watertight unit.

In order to give support to the inner ends of the DB (down beam), HR-3, HR-4, HG-5, HG-6, and HG-9 (horizontal graphite) liners, sockets are welded to tank section *D* at appropriate places. Some of these are shown in Fig. 3-8.

Top and bottom plugs. The top and bottom plugs complete the watertight shell around the active lattice and form part of the support system for the control rods.

The bottom plug, shown in Fig. 3-9, is a hollow stainless-steel shell filled with lead shot to fulfill the shielding requirements under the reactor. Its total weight is approximately 6 tons, and while it can be removed, only the direst necessity would justify such a step. Through the bottom plug pass the discharge chute, the hydraulic rabbit (shuttle) tubes, and the monitor tubes (see later sections for description of these). This plug also supports the shim-rod lower shock absorbers (see Sec. 3-6).

The top plug (see Fig. 3-10) is not only the upper tank closure but also is the primary support for the control rods and their drive mechanisms. Like the bottom plug, it is a hollow stainless-steel shell partially filled with lead shot to meet shielding requirements. On this plug is mounted a platform which supports the control motors, limit switches, etc., for the control rods. The spider which holds the bearings for the 12 control rods (see Sec. 3-5) hangs underneath the top plug on four supports.

Since the top plug must be taken off in order to replace fuel elements or control rods, it is designed to be lifted easily by the overhead crane and lowered to a "dry dock" on the first floor.

3-5 Active Lattice. The fuel for the reactor is enriched uranium alloyed with aluminum and made up into aluminum-clad plates 0.060 in. thick. Eighteen of these plates, each approximately 2.8 in. wide and $24\frac{5}{8}$ in. long, are made into a single fuel assembly. The plates, with a 0.118-in. space

between them, are brazed into aluminum side plates, and the assembly is then equipped with aluminum end boxes. A cross section of the assembly is shown in Fig. 3-11, and a picture of completed fuel pieces is shown in Fig. 3-12.

Referring to Fig. 3-12, the bottom end of the fuel assembly is the end without the spring. Both

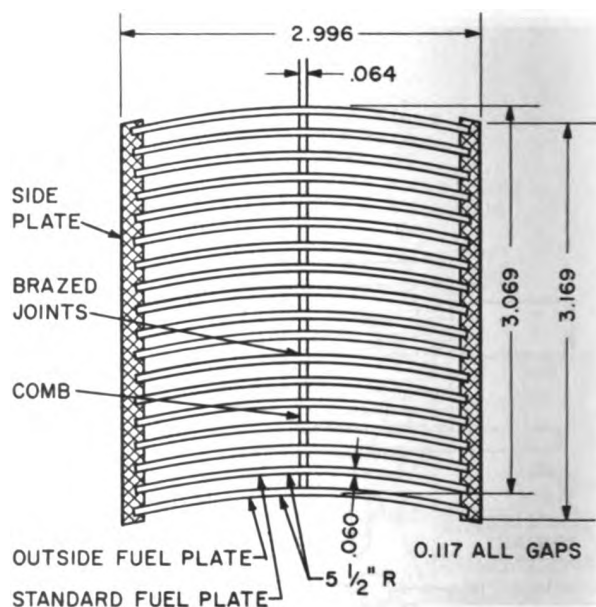


Fig. 3-11 Cross section of fuel assembly.

end boxes are, of course, open to allow water to pass freely through the assembly from top to bottom.

The arrangement of the fuel assemblies into a lattice and the supporting structure of the assemblies can best be seen in Figs. 3-8 and 3-9.

Referring to Fig. 3-9, it will be seen that the end boxes of the fuel assembly fit into the upper and lower assembly grids. The spring section of the upper end box (see Fig. 3-12) allows each assembly to be held firmly and still allows for expansion and tolerance limitations. The eight square holes in the upper assembly grid are for the shim-safety rods which extend all the way through the reactor.

The actual assembly of all these parts is best seen in Figs. 3-8 and 3-9. The lower support casting, which is bolted to tank section *D*, supports the lower assembly grid and the lower bearing grid. The upper support casting supports a re-

movable assembly of the upper assembly grid, the grid spacer, and the upper bearing grid. This assembly can be locked in place or lifted as a unit by the upper locking mechanism.

A further set of control-rod guide bearings, mounted in the spider, is fastened to the top plug by

Since the control rods must operate freely, it is essential that the lower shock absorbers mounted on the bottom plug and all bearings in the various grids line up accurately. The lower shock absorbers are adjustable as a group. Final adjustment of the shock absorbers and the doweling of

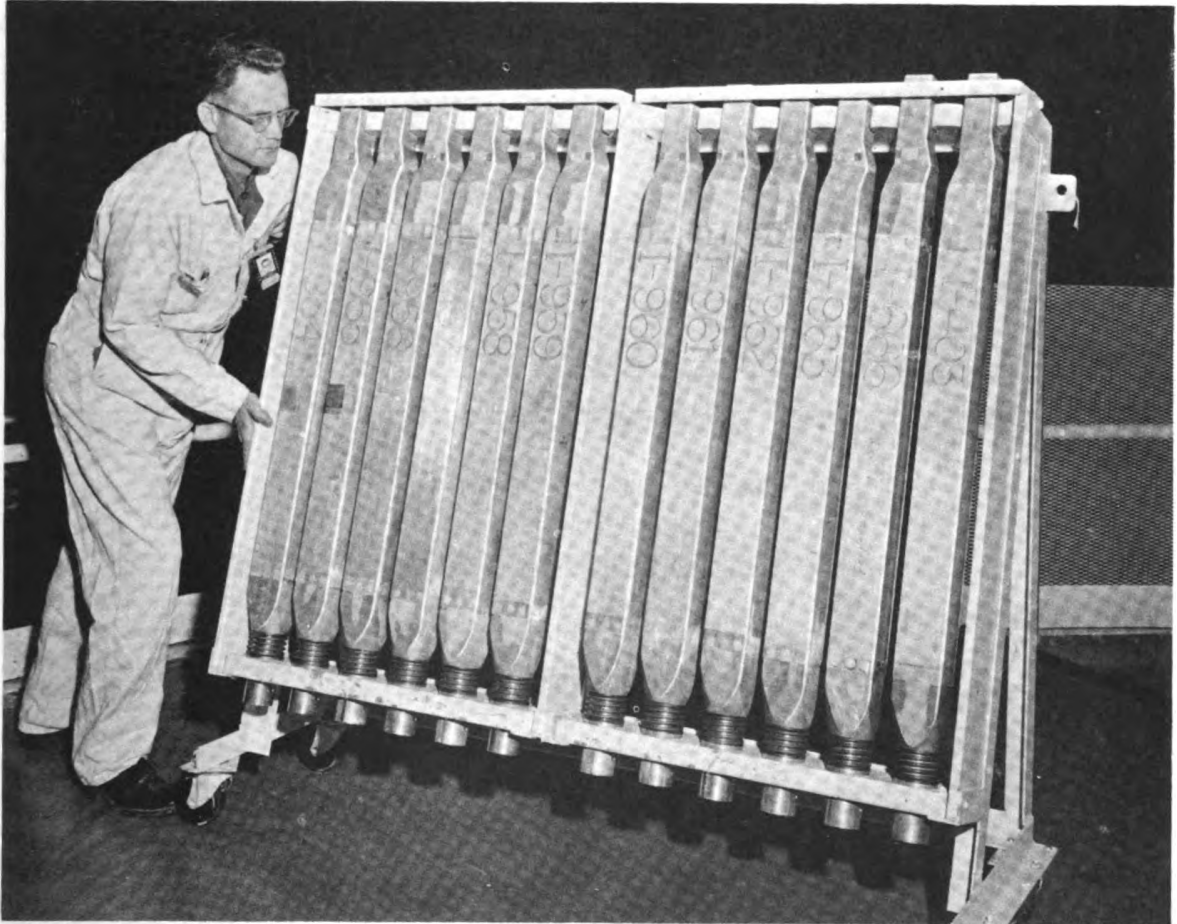


Fig. 3-12 Fuel-assembly units.

four long support tubes. When the top plug is lowered, the four corners of the spider fit over pins on the spider support ring, which then holds the spider in the correct position.

In order to allow for machining tolerances and slight errors in lowering the top plug, the spider is actually hung on the support rods by rather light springs. In this way, although the bearings and spider are attached to top plug, they are aligned by the rigid spider support ring which is bolted to the top flange of tank section *B*.

the support grids and top plug were done during construction.

Once the system is aligned, fuel is loaded by means of special handling tools in the following manner: With the top plug off, the upper grid assembly is lifted from the upper support casting and placed on a special support rack mounted on the spider support ring. The shim-safety rods are then placed in position, being supported by the lower grids and the shock absorbers. Finally, the fuel assemblies are lowered into the lower

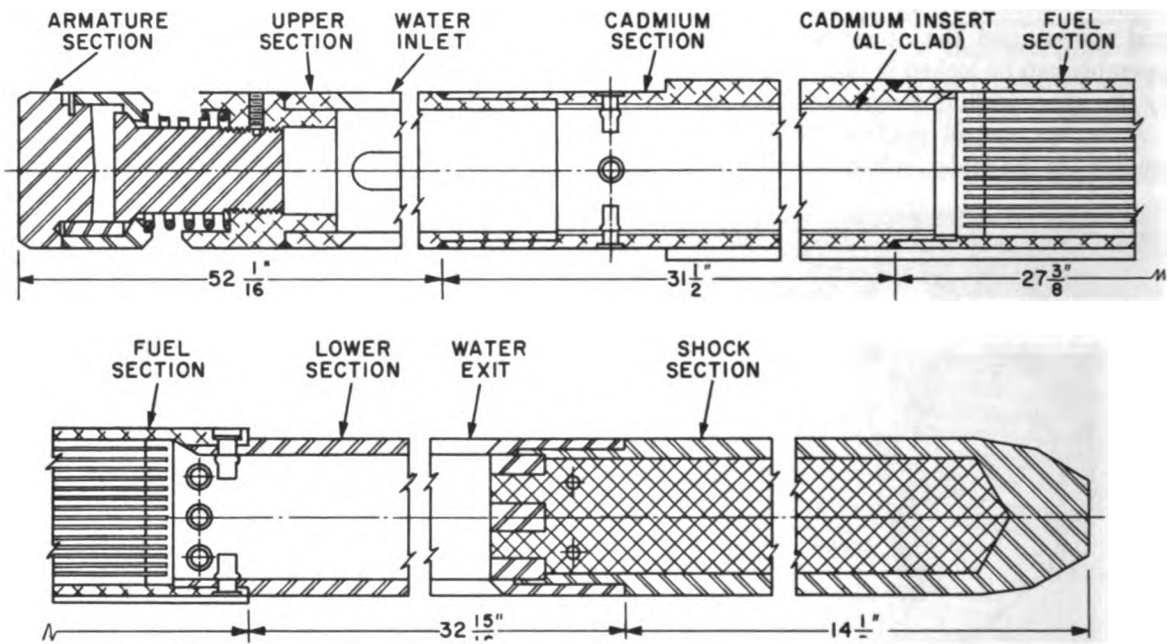


Fig. 3-13 Shim-safety rod with fuel.

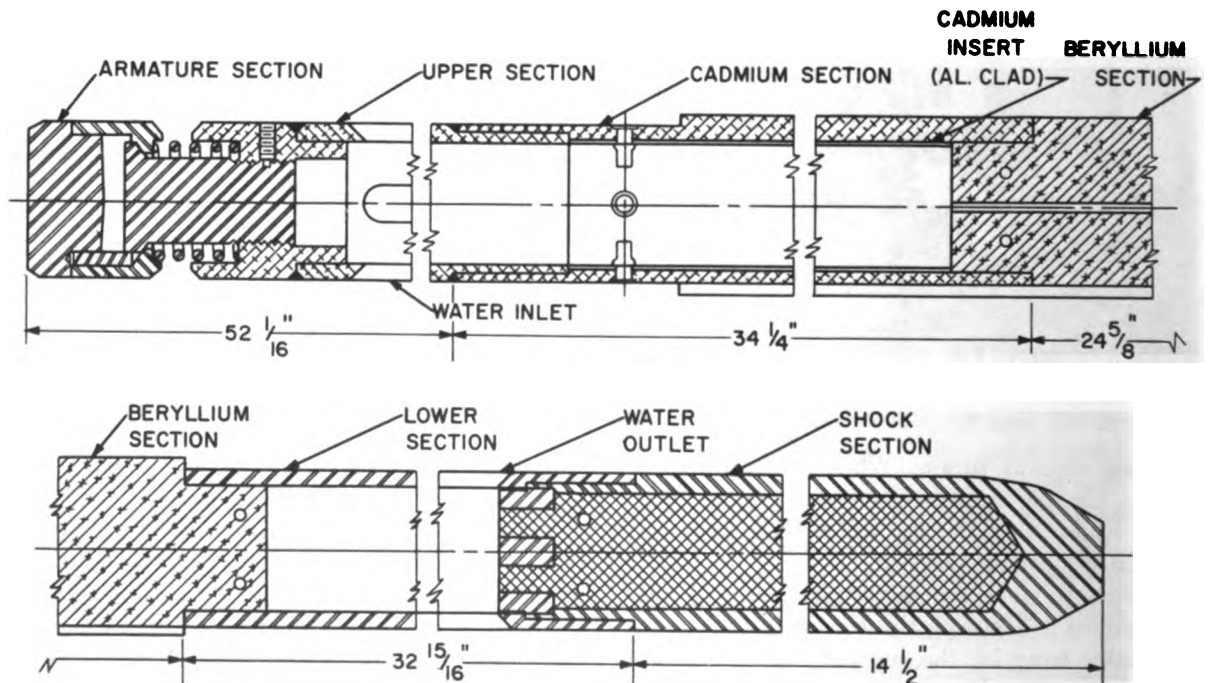


Fig. 3-14 Shim-safety rod with beryllium.

assembly grid one at a time, with due regard to criticality (see Figs. 3-13 to 3-15). When enough fuel is loaded, empty spaces in the lattice are filled with beryllium pieces (L pieces) of the same shape and size as the fuel elements. When this operation is completed, the upper grid assembly is lowered over the top of the fuel element end boxes and clamped down, the control rods at the same time

experimental holes HB-1, HB-2, HB-3, DB-1, DB-2, and DB-3 receive a higher fast flux than the holes on the opposite side which "see" the lattice through two additional rows of beryllium.

The reactor is unloaded by lowering the fuel elements through the discharge chute into the canal. An outline of the procedure is as follows: With the shim rods and regulator rods discon-

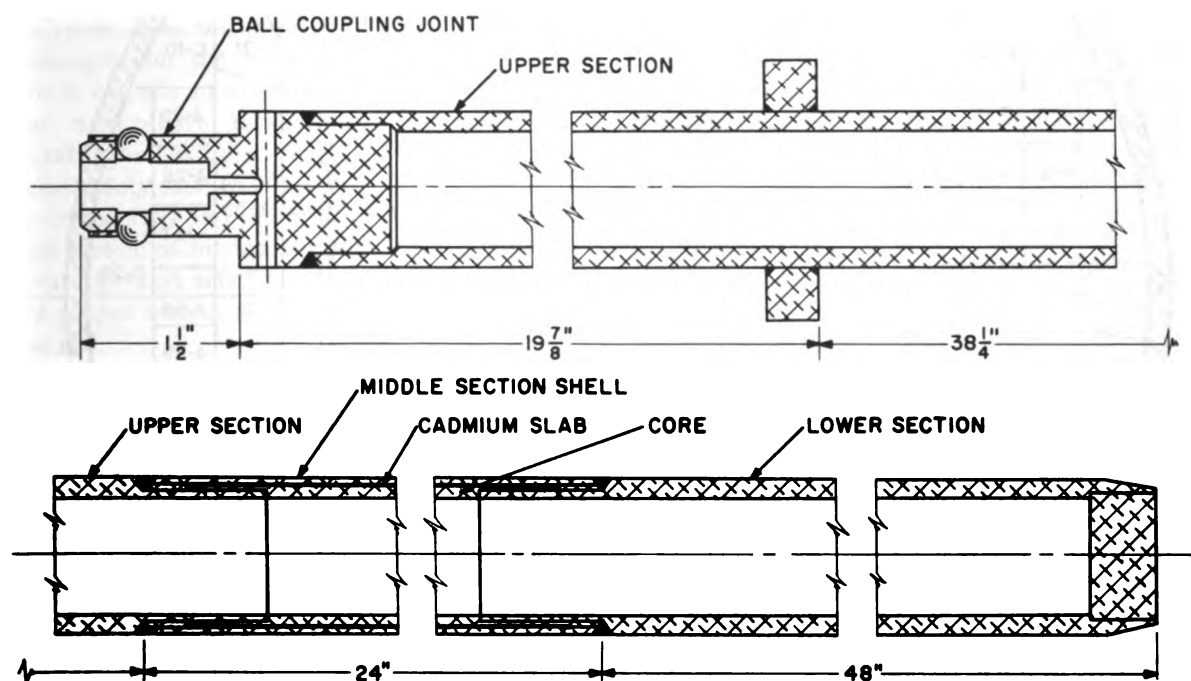


Fig. 3-15 Regulating rod.

extending through the upper bearing grid. The top plug, now holding the upper halves of the control rods extending through the spider, is then lowered and bolted down.

It was planned to use a narrow slab loading, nine assemblies long by three wide. The assembly grids allow 45 places (including shim-safety rods) with the long side of the grids oriented in the east-west line. The lattice was initially loaded on the north side, thus putting one row of four shim-safety rods in the central line of fuel assemblies. In keeping with this planning, it will be noted that HG-9 is placed off center as far as the structure is concerned but "looks" at the center of the loaded lattice. In this arrangement,

nected and resting in their lowest position, the top plug is unbolted, lifted, and carried by the crane to the dry dock. The upper grid assembly is then lifted from its support casting and placed on the support rack on the spider ring. Before any fuel elements are lifted, the beryllium pieces D-1 and D-2 (see Fig. 3-16) are lifted and placed to one side of tank section D. Then the discharge-chute plug is lifted, leaving the discharge chute open to the receiver cylinder in the canal. A fuel assembly or control rod is then raised clear of the active lattice, moved sideways to the discharge chute, and lowered to the receiver cylinder in the canal. Removal from the receiver into the canal is fully described in Secs. 3-17 to 3-21.

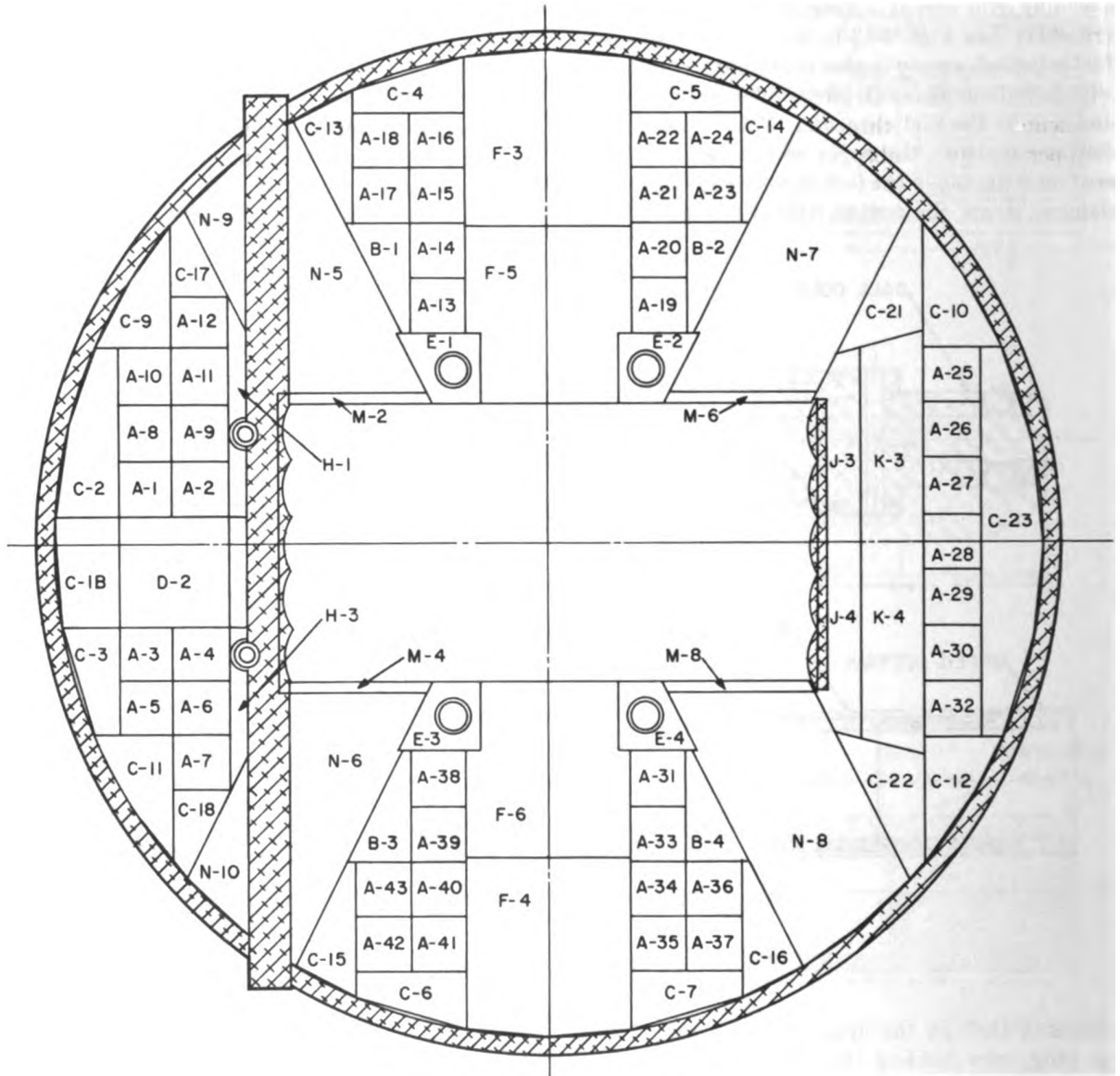


Fig. 3-16 Beryllium below center line of reactor.

3-6 Control Rods. The MTR control rods are of two main classes, the shim-safety rods and the regulating rods. All of them are designed to lower the reactivity of the reactor when they are inserted, and when fully in, i.e., resting in their shock absorbers, they will overcome an excess reactivity of greater than 40 per cent. Since it is necessary to have approximately 19 per cent excess reactivity built into the reactor, the control rods provide an adequate margin of safety.

Shim-safety control rods. The shim-safety control rods, as their name implies, serve the purpose of providing both regulation and safety control. It has already been noted that there are places for eight shim-safety rods running through the active lattice. At present there are two types of rods. Four of them are constructed to bring fuel sections, each containing 14 fuel plates, into the lattice as they are raised and are used in the four positions in the middle of the fuel slab. The re-

maining four have beryllium in the lower section, since they are in the beryllium section of the lattice grid. All eight have cadmium sheet in the upper section, and this section will reside in the lattice when the rods rest in the lower shock absorbers.

The two types of shim-safety rods are shown in Figs. 3-14 and 3-15.

The rods as shown extend from the shock absorbers up through the upper bearing grid. They are raised from this position by means of an electromagnet on the bottom of the lifting shaft, which extends upward through the top plug to the motor-drive mechanism. This magnetic clutch provides the "scram" or emergency-shutdown safety feature of the rods. Its operation is described fully in Secs. 3-22 to 3-32. The armature section at the top of the safety rods contains a spring which acts like a universal joint so as to aid proper seating of the magnet onto the top of the rod.

Under normal operation, the shim-safety rods are driven by means of the magnetic clutch and vertical lifting shaft by reversible three-phase induction motors and worm drives (see Fig. 3-10). The motors are remotely controlled by reversing contactors, suitably interlocked and having the usual complement of upper and lower limit switches. The advantage of the induction motor is that no accident can make it run in excess of synchronous speed; therefore, the danger of over-speed withdrawal of the shim-safety rods is eliminated. No brake is required since the worm and screw drive is self-locking.

Pertinent data on the shim-safety rods are summarized in Table 3-1.

Table 3-1 Cadmium-Uranium Shim Rod

Weight of rod.....	110 lb
Weight of magnetic clutch.....	40 lb
Weight of drive shaft.....	35 lb
Total weight of rod, shaft, and clutch.....	185 lb
Water-pressure load on rod, assuming approximately 40 psi.....	290 lb
Drive motor power.....	$\frac{1}{4}$ hp
Drive motor speed.....	1725 rpm
Water-flow area (upper section)...	5.06 in. ² (32.6 cm ²)
Water-flow area (uranium section)	4.2 in. ² (26.1 cm ²)
Water-flow area (lower section)...	5.06 in. ² (32.6 cm ²)
Water-flow area (cadmium section)	5.04 in. ² (32.4 cm ²)

Regulating rods. The regulating rods are designed to provide continuous control of the reactor. Although four locations are provided for these rods in the beryllium reflector (see Fig. 3-16), only one is normally used for control, with one more as a stand-by. The remaining two locations are used for hydraulic rabbit (shuttle) tubes as experimental facilities.

As shown in Fig. 3-15, each rod is $1\frac{1}{2}$ in. in diameter and consists of an upper aluminum section, a middle section containing cadmium between aluminum tubing, and a lower aluminum section. Guide bearings are provided in the upper and lower support castings. The regulating rod itself is connected by a quick-release mechanism to the regulating-rod support shaft, which is guided by bearings in the spider and the top-plug platform (see Fig. 3-17).

When connected to its support shaft, each regulating rod is driven by a low-inertia electric motor through a pinion and rack integral with the shaft. The motor is driven by an amplidyne, as described in Secs. 3-22 to 3-32.

When disconnected, the regulating rod rests on the top of the upper bearing grid. In this position the lower tip of the regulating rod and the lower tip of the cadmium section are 60 and 12 in., respectively, below the center of the pile lattice. Table 3-2 summarizes the pertinent data for the regulating rods.

Table 3-2 Regulating Rod

Composition	Cadmium-Aluminum
Weight of regulating rod.....	10.5 lb
Weight of drive rod.....	36.0 lb
Weight of water in drive rod.....	9.0 lb
Total weight of combined rods.....	55.5 lb
Buoyant force of regulating rod.....	8.1 lb
Buoyant force of drive rod.....	9.8 lb
Total buoyant force.....	17.9 lb
Resistance force of rod through pile section (friction).....	7.4 lb
Acceleration.....	1.3 g
$\Delta k/k$ per rod.....	0.005
Drive motor rated power.....	1 hp

Miscellaneous equipment connected with control rods. **Drive shafts.** The drive or support shafts for the control rods provide primarily a connecting link between the rods and their driving mechanism. The drive shafts are in three sec-

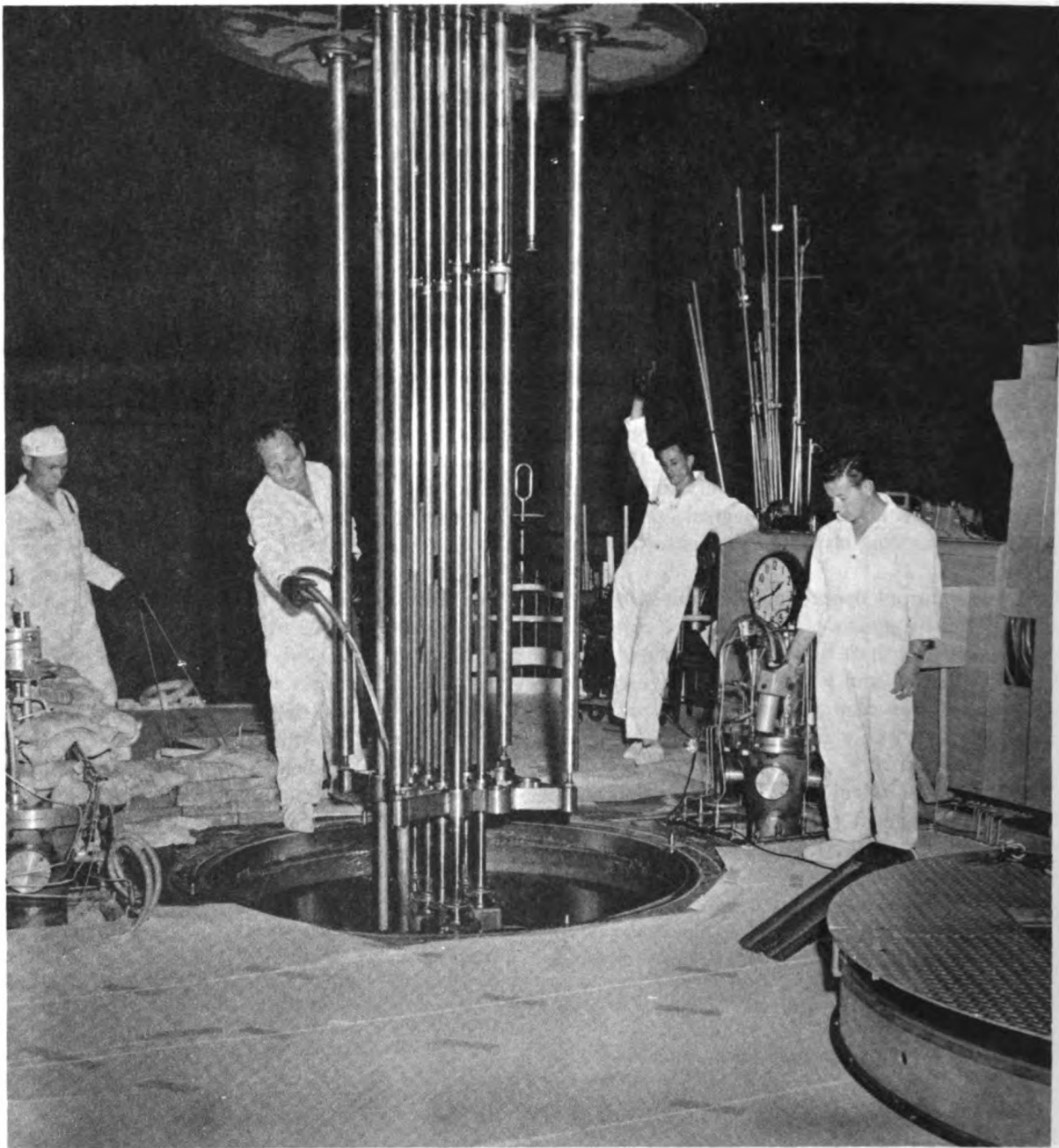


Fig. 3-17 Removing rod assembly from reactor tank.

tions. The top section of stainless steel carries the rack with an acme thread to fit the driving mechanism. The two lower sections are of aluminum, the middle piece being filled with lead shot to reduce gamma radiation; the bottom section carries the electromagnet in the case of the shim

rods and the quick-release mechanism in the case of the regulating rods.

Upper and lower rack housing and cover, regulating rod. The upper rack housing, fabricated from cast iron, is fastened to the upper surface of the top-plug platform (see Fig. 3-10). It functions as

part of the guide and bearing for the regulating-rod drive rack, houses the rack pinion, and locates the regulating-rod speed reducer and position indicator. The rack-housing cover is bolted to the housing and completes the regulating-rod guide and bearing. Microswitch brackets are fastened to the rack-housing cover. Five of them are located on one side of the vertical center line and two on the other side. One of the latter, the long bracket, accommodates two microswitches, which are adjustable with respect to each other. These microswitches are used in the interlock system which controls the relative motions and positions of the regulator and shim rods. This system is fully discussed in Secs. 3-22 to 3-32.

The lower rack housing, attached to the under-surface of the top-plug platform, functions as a seal for the regulating-rod drive shaft and as a support for the lower shock absorber.

Upper and lower shock absorbers, regulating rod. The upper shock absorber is bolted to the upper surface of the upper rack housing and cover. It dissipates the energy in the motor when the amplidyne armature circuit fails to stop the servomotor on the regulating-rod upstroke. It also serves as the upper end housing for the regulating-rod drive. The shock absorber is capable of resisting a force greater than 1300 lb, occurring when the rod moves upward with a velocity of 4 fps.

The upper end of the drive rack engages a plunger and moves it against an oil cushion, forcing the oil through an annulus between a flange on the plunger and the tapered wall of the housing. The oil is retained in the chamber formed between the housing and the guide by means of O-ring seals. A spring returns the plunger to its normal position. A threaded plug is provided in the top of the shock absorber to permit access to the rod-locking device.

The lower shock absorber is fastened to the lower rack housing and is similar in principle to the upper shock absorber. It dissipates the energy in the servomotor when the rod is driven to its lowest position.

Drive-rod top-plug seal. The top-plug seal for both the regulating and shim drive rods is designed to prevent water leakage where the drive rods pass through the top plug. A packing nut, when tightened against its seat in the seal housing, holds chevron leather packings against the wall of

the seal housing and the drive rod. As an added precaution against the leakage of contaminated water from the reactor, a purging system is incorporated into the design. This is accomplished by introducing a supply of fresh water under the seal at a pressure slightly greater than the water in the reactor tank. The water flows through the annulus between the regulating-rod drive and its bearings into the reactor tank.

Position indicators. The regulating- and shim-rod position indicators have been designed to indicate the exact position of the rods at any time. Rod positions may be read by means of direct-reading dial indicators at the platform of the top plug and by means of remote selsyn-driven (dial) indicators at the console in the control room.

The regulating-rod primary indicators at the top-plug platform are double-disk dials operated by means of suitable gearing. The outer disk shows the position of the rod in inches, and the inner disk indicates to hundredths of an inch. One complete revolution of the outer disk provides indication for 30 in. of travel of the rod, while the inner disk makes one revolution for each inch of travel of the rod.

The secondary position indicator at the console is a dial pointer which moves 240° for 12 in. of travel of the regulating rod. This means that 20° of rotation of the pointer is equivalent to an inch of rod travel.

The shim-rod primary indicators at the top-plug platform are standard counters with five digits driven through suitable reduction gearing attached to a flexible shaft, which in turn is driven directly from the shim-safety-rod drive motor. These indicators will show the position of the rod to thousandths of an inch.

The shim-rod secondary position indicator at the console is a dial pointer which rotates 9° for each inch of rod travel; thus for a total travel of the rod of 31 in., the dial will rotate 279° .

The secondary indicator in both instances is operated by means of a receiver selsyn. The regulating-rod-driver selsyns are located at the regulating-rod driver, while those for the shim rods are located under the top-plug platform. All are driven by means of suitable gearing, and in all cases either the primary or secondary indicator may be removed for servicing without affecting the other.

Speed reducer and motor coupling, shim-safety rod. The speed reducer is essentially a helical gear and pinion supported by ball bearings, mounted in a housing which bolts to the upper rack housing. The position indicator operates from a bevel gear fastened to the helical gear shaft. Gears and bearings are lubricated by immersion in oil. An oil seal, pressed into the pinion bearing cap, prevents oil leakage around the pinion shaft.

The motor coupling is the connection between the speed-reducer shaft and the motor shaft. It is a splined collet which fits the spline shaft of the motor and is fixed by a taper pin to the speed-reducer drive shaft. The taper pin acts as a shear pin if an excessive load is placed upon the shim rod. The collet is threaded and tapered for an adjustable nut. When the nut is tightened, it clamps the collet to the splined motor shaft, thus eliminating any slipping between motor and speed-reducer shaft.

Magnetic clutch. This clutch is essentially a horseshoe or bipolar electromagnet. The pole cores are made of laminated SAE 1010 steel, $\frac{1}{16}$ -in. laminations insulated with baked-on Bakelite varnish 0.002 in. thick. The energizing coils have 3500 turns of No. 25 Formex-coated copper wire, wound on forms of Fosterite Vacuum Impregnate.

The clutch housing is grounded and forms one side of a circuit which indicates when the magnetic clutch is supporting the shim-safety rod. The casing is a watertight can of type 304 stainless steel, with welded seams, gasketed at the top with polyethylene. The bottom is flanged and tapped for a diaphragm sealing flange. Nonmagnetic shim material is a single sheet of stainless steel 0.008 in. thick and a sheet of mica 0.002 in. thick.

The oil filling used for this clutch is perfluorodimethylcyclohexane. Pressure build-up due to gassing of the oil is avoided by a vent to atmosphere through the cable channel.

On the basis of previous test data, the holding force of this clutch should be a minimum of 500 lb at 100 ma current, and it should release a 600-lb load held at 140 ma current in 15 msec or less.

Bearings. Each shim-safety rod is guided by two bearings consisting of four spring plates, in

each of which there are mounted three graphite rollers encased in stainless steel. These plates are mounted in a steel cage and are located in the upper and lower guide grids. The center of the upper bearing is located 3 ft 10 $\frac{5}{8}$ in. from the center line of the reactor and is supported by the upper grid support. The center line for the lower bearing is located 4 ft $\frac{3}{4}$ in. below the center line of the reactor and is supported from the lower grid support.

Bottom absorber, shim-safety rod. The bottom shock absorbers for the shim-safety rods are located on a plate mounted on the bottom plug. They are merely dashpots into which plungers mounted on the bottom of the shim rods fit. Since they are on the bottom plug, they are always filled with water, which acts as the damping fluid.

These shock absorbers also contain a mechanism which indicates whether or not the rods are seated in the shock absorber and are therefore fully in the reactor. This is accomplished by introducing a high-pressure demineralized-water line through the bottom plug into each shock absorber. The pressure in this line will be maintained above that normally at the bottom of tank section *E*, so that when the shock absorber is not seated water flows up into the tank. When the rod is seated, however, the flow is decreased and appropriate signals from the pressure meters inform the reactor operator of the situation.

3-7 Reflector. Beryllium. The beryllium section of the MTR reflector is entirely contained inside tank section *D* (see Fig. 3-16), occupying the space between the active lattice and the tank wall. This beryllium has a finished weight of about 5750 lb and a volume of nearly 50 ft³. The total height of the reflector is 39 $\frac{3}{8}$ in. (approximately 100 cm); the layout and shape of the pieces are shown in Fig. 3-16. These diagrams are cross sections below and above the beam holes, and they show that the majority of the pieces are continuous throughout the 100-cm length.

A brief description of the various pieces shown in Fig. 3-16 follows. The key pieces are those numbered E-1 to E-4, which contain large holes throughout their length for passage of the regulating-rod bearing mount in the lower grid (see Fig. 3-8). The A pieces (A-1 to A-43) are fitted

with "end boxes" which fit tightly into the holes in the lower support grid (see Fig. 3-9). Tolerances allow an average spacing of 0.020 in. between pieces. The remainder of the pieces are large heavy blocks, machined to fit into their respective places, and are not rigidly held. The water film between surfaces prevents surface abrasion, and appropriate holes are provided for the passage of cooling water. For example, each A

into the lattice (see Sec. 3-15). The packing is somewhat complicated by the aluminum bar which contains the holes for the horizontal pneumatic rabbits (see Secs. 3-10 to 3-14). This bar is shown between N-5 and N-9 and between N-6 and N-10 in Fig. 3-16.

While the beryllium pieces C-1-A and C-1-B, as well as D-1 and D-2, are shown as separate pieces, they are mechanically joined by a sealed

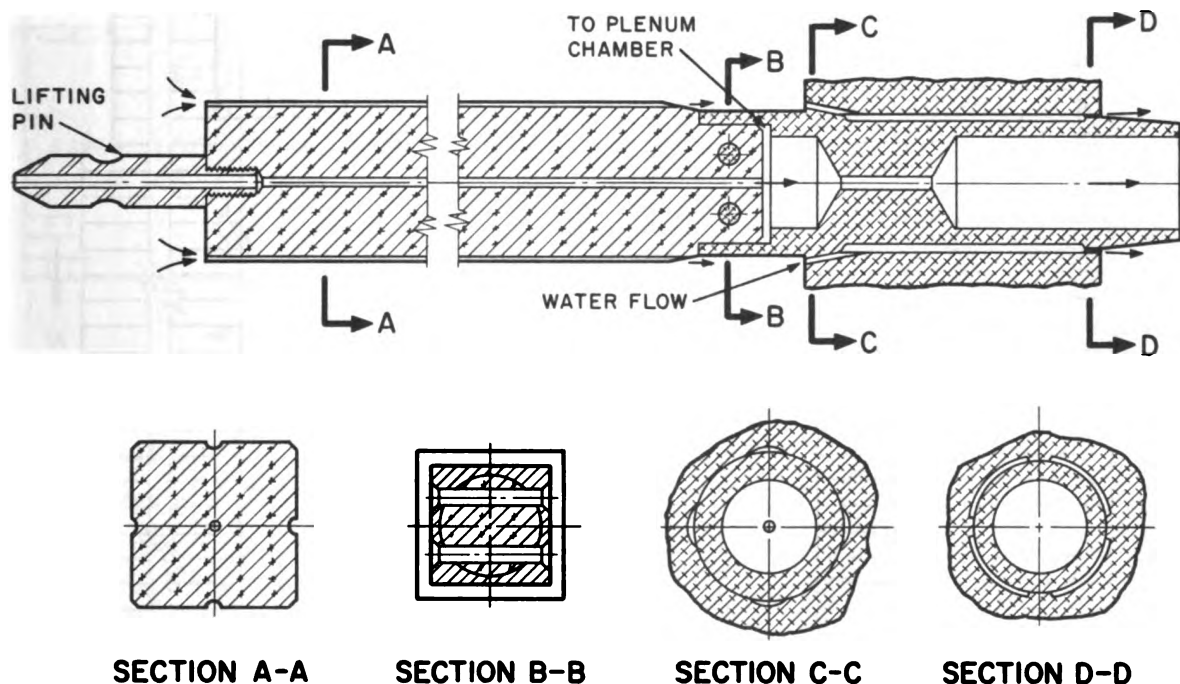


Fig. 3-18 Beryllium A piece.

piece has an axial hole $\frac{3}{16}$ in. in diameter and a semicircular groove down each face $\frac{1}{8}$ in. in diameter (see Fig. 3-18). The axial $\frac{3}{16}$ -in. hole was necessitated not by the volume of cooling water required but by the surface necessary for heat transfer and by the surface stress set up by heat differentials in the beryllium.

The large pieces N-1, N-2, N-3, N-4, F-1, and F-2 are semicircular on the bottom to fit over, but not touch, the HB thimbles. These large pieces actually rest on the lower beryllium pieces N-5, etc., which are grooved on top to fit under the respective thimbles. Plates M-1 to M-8 are fastened to the N pieces to provide a narrow tight-fitting collar around the inner end of the thimbles. This prevents water leakage from the reflector

empty aluminum can which effectively eliminates the reflector from hole HG-9. D-1 and D-2 form the plug for the discharge chute and are lifted out as one piece when it is necessary to use the chute.

Graphite. Part of the purpose of the MTR is to provide as large a neutron flux as possible over as great a volume as possible. To accomplish this, a secondary reflector of graphite is provided outside the main reactor tanks.

The main problem faced in the design of the graphite reflector was the accommodation of any graphite expansion due to radiation or temperature effects, or to both. To overcome this difficulty the graphite reflector is composed of two zones: a pebble zone, which is adjacent to the reactor tank and extends for a minimum of 40 cm

from the tank wall to form a square; and a permanent graphite zone 12 ft over all north and south, 14 ft over all east and west, and 9 ft 4 in. high (see Figs. 3-3 and 3-4).

The pebble zone is a volume 7 ft 4 in. square

plate can be opened into a discharge bin to allow removal of the pebbles at any time. This very easy removal of the pebbles carries out the MTR feature which provides for removable *B*, *C*, and *D* tank sections (see Sec. 3-4).

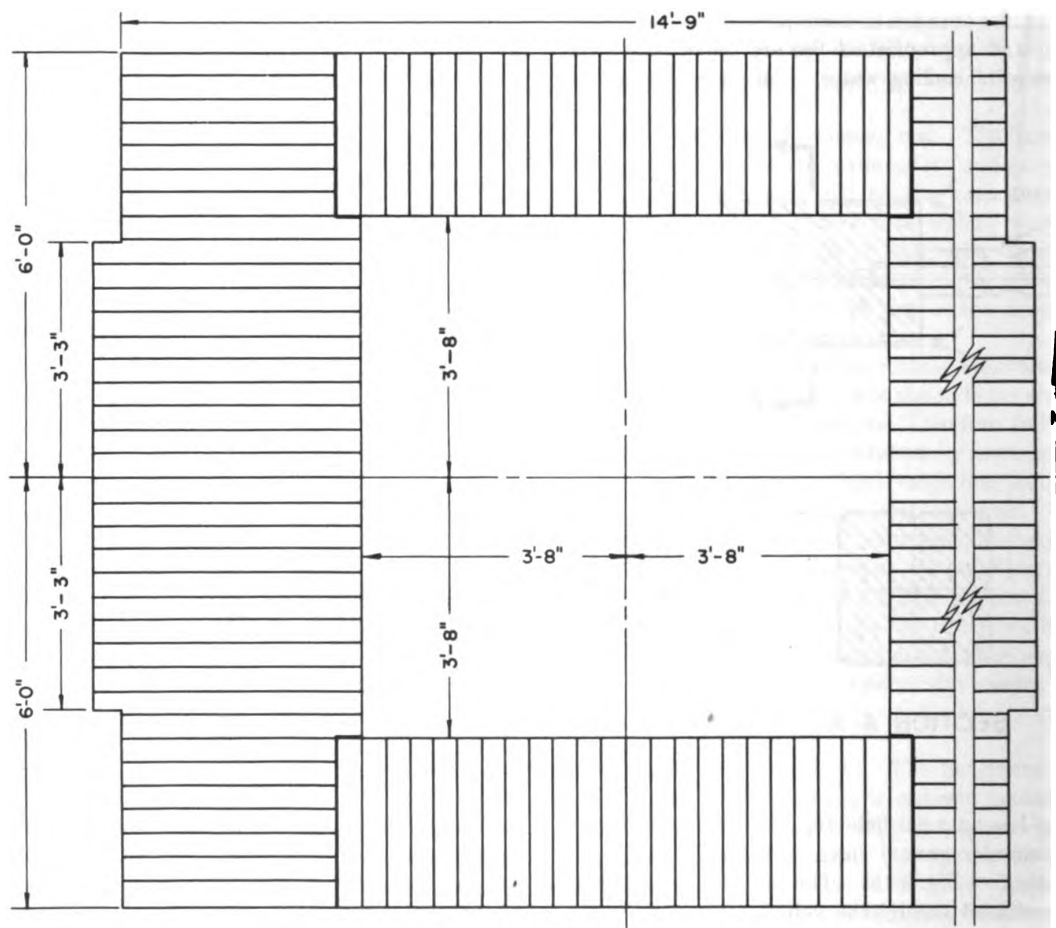
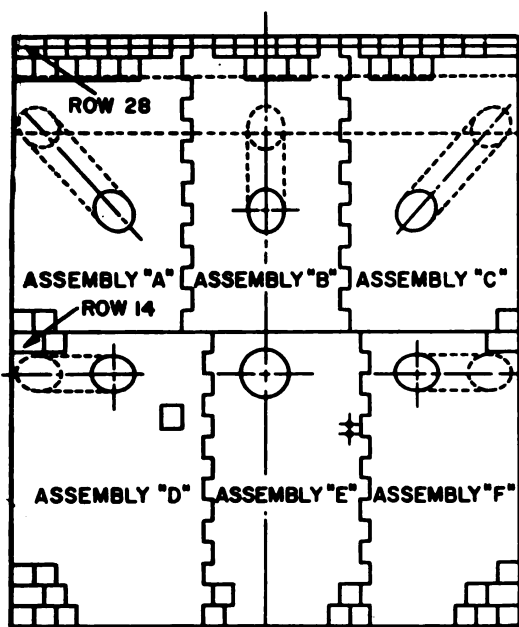


Fig. 3-19 Permanent graphite moderator.

and 9 ft high surrounding the reactor tank and filled by about 700,000 graphite pebbles 1 in. in diameter. The pebbles rest on a base plate mounted on the lower thermal shield plate and are free to expand upward. In this way any expansion will only raise the level of the balls without exerting undue stress on the tank wall or the permanent graphite.

Two discharge chutes in the pebble support

The permanent graphite is composed largely of 4- by 4-in. graphite bars stacked to a height of 9 ft 4 in. with the bottom located 4 ft below the center line of the active section (i.e., at elevation 96 ft; see Sec. 3-4). The system of stacking the graphite is shown in Figs. 3-19 and 3-20. The first of these figures shows the over-all relationship of all four walls at one level, while the second figure illustrates the stacking and numbering sys-



KEY TO MATCH-MARKING SYSTEM

INDIVIDUAL BLOCKS ARE MATCH-MARKED AS SHOWN IN SAMPLE ASSEMBLY "A" AND DESCRIBED BELOW :-

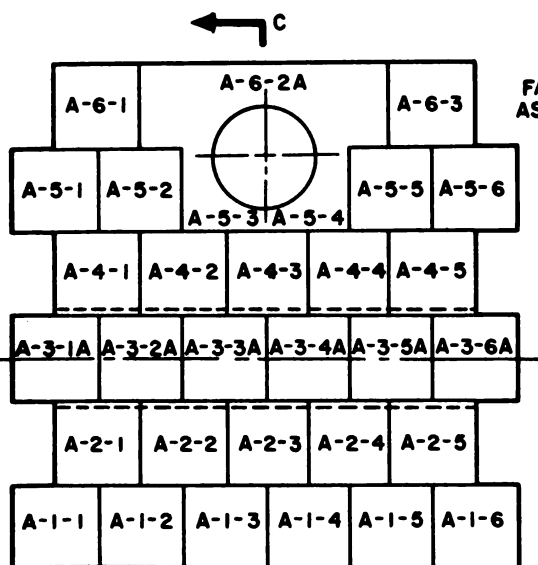
1st SYMBOL (LETTER) - ASSEMBLY IN WHICH BLOCK APPEARS.

2nd SYMBOL (NUMBER) - ROW IN THE ASSEMBLY IN WHICH BLOCK APPEARS.

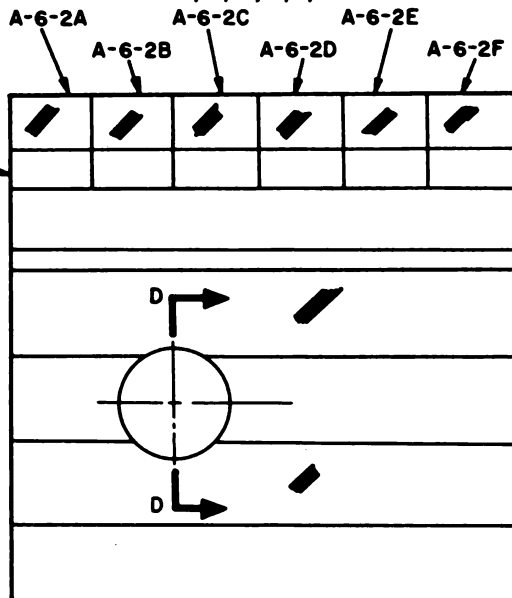
3rd SYMBOL (NUMBER) - POSITION OF THE BLOCK IN THE ROW; NUMBERS STARTING FROM LEFT SIDE OF THE ASSEMBLY, VIEWED FROM CENTER OF REACTOR.

4th SYMBOL (LETTER) - POSITION OF THE BLOCK FROM FRONT FACE OF ASSEMBLY. LETTERS STARTING FROM FRONT FACE OF ASSEMBLY. (SEE SECTIONS C-C AND D-D.) THIS SYMBOL IS USED ONLY WHERE ONE BLOCK DOES NOT EXTEND THE FULL DEPTH OF THE ASSEMBLY.

WEST WALL ASSEMBLY A, B, C, D, E, F
EAST WALL ASSEMBLY G, H, J, K, L, M
SOUTH WALL ASSEMBLY N, P, Q, R, S, T
NORTH WALL ASSEMBLY U, V, W, X, Y, Z

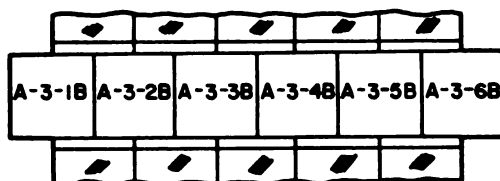


FRONT FACE OF ASSEMBLY



SECTION C-C

SAMPLE ASSEMBLY "A"



SECTION D-D

Fig. 3-20 Graphite assembly and numbering system.

tem. It will be noted that every other row has a block 4 by 6 in. at each end to allow for the staggering of the rows.

The stacking was, of course, complicated by the experimental holes penetrating the permanent graphite. These larger holes were cut into the walls by clamping a group of bars together and drilling through the assembly. Each wall array is made integral by a system of keys 1 by 1 in. in size and the entire assembly is keyed at its inner face to the supporting rails. Soft asbestos fills the gap between the permanent graphite and the thermal shield.

Each wall of the permanent graphite is mounted on rails running parallel to its length. The rails are supported by jackscrews which are fastened to the lower plate of the bottom thermal shield and pass through clearance holes in the inner plate of the lower thermal shield (see Figs. 3-3 and 3-4). The support rails must be accurately leveled in order that the graphite will stack properly and be in correct location for the beam holes.

High-purity graphite is, of course, desirable because of lower induced activity and lower neutron absorption. The final graphite reflector and thermal column consist of the following:

Approximate Finished Weight, tons

Pebbles.....	19
Permanent zone....	74.5

The graphite reflector, owing to the heat absorbed in it, must be cooled. This is done by drawing large quantities of air through holes in the permanent graphite and through the interstices between the balls in the pebble zone. This cooling system is fully described in Sec. 3-16.

3-8 Thermal Shield. *The steel-plate structure.*

The thermal shield of the materials-testing reactor is that part of the reactor structure between the solid graphite zone and the concrete biological shield. It has been established that a 50°F temperature drop across the concrete biological shield is the temperature differential which can be tolerated without danger of failure within the concrete structure due to induced thermal stresses. The MTR thermal shield is a large steel structure which reduces the neutron and gamma flux to

such an extent that a comparatively low heat generation due to radiation occurs in the concrete.

The thermal shield, consisting of steel plates 4 in. thick, may be visualized in so far as orientation with respect to the rest of the reactor is concerned by referring to Figs. 3-4 and 3-5, which are vertical and horizontal sections through the reactor, respectively.

The approximate over-all size is as follows: 16 ft in the north-south direction, 14 ft in the east-west direction, and 12 ft 6 in. in height. The thermal shield, whose heat-generation rate is approximately 95 kw, is cooled by an induced-draft air system which is also the graphite cooling system (see Sec. 3-16). The primary supporting structure is four vertical WF columns with the load transmitted to these columns at the four corners of the bottom plate.

The bottom plate of the lower thermal shield, although resting directly on the steel supporting structure, is in direct contact with the barytes concrete below it. This helps to cool the concrete by direct thermal contact. The upper plate of the lower thermal shield is in two parts. An inner ring 14 in. wide is welded directly to tank section *E* (see Fig. 3-9) to provide maximum cooling for this section. The outer section is supported from the lower plate and is air-cooled only.

The solid graphite is supported from the lower plate of the bottom thermal-shield plate (as described in Sec. 3-7). The two plates of the upper thermal shield are independently supported from the concrete above them, and with the exception of one point, which is a metal expansion joint air seal, the top plates do not touch the side plates of the thermal shield. Figure 3-21 shows the top-plate arrangement. All the eight side plates rest on the lower plate of the bottom thermal shield and are essentially independent units. The four inner plates, and similarly the four outer plates, are loosely connected at the top corners by dumb-bell-shaped pins which fit in oversize slots in the tops of adjacent plates.

Two important experimental facilities, the thermal column and the shielding facility (see Secs. 3-10 to 3-14), require that a section approximately 6 by 6 ft be removed from the east and west thermal-shield side plates. Two components are added to the thermal-shield design by these facilities: (1) the neutron window and (2) the neutron-

absorbing curtain. The locations of these components may be visualized by referring to Figs. 3-4 and 3-5. When the 6- by 6-ft hole was cut in the thermal shield, it became necessary to replace the steel removed on the thermal-column side by a material that absorbs gamma radiation impinging on the inner side of the thermal shield and yet allows passage of thermal neutrons into the thermal column. The lead that replaces the steel in this region is called the "neutron window."

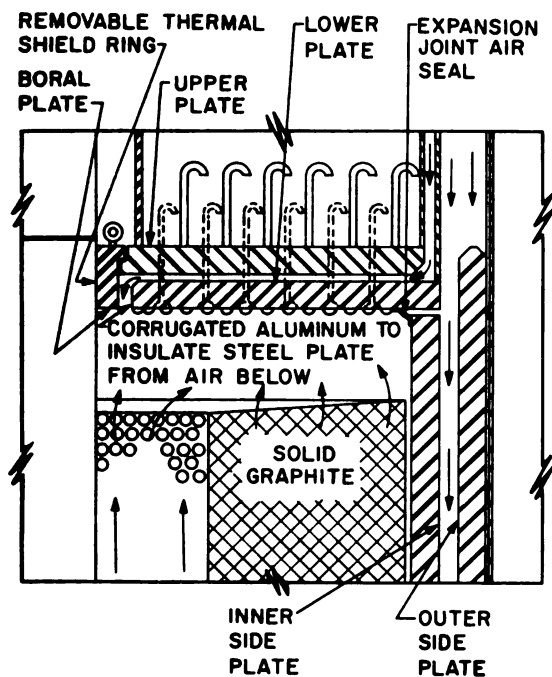


Fig. 3-21 Top thermal shield design.

The "neutron-absorbing curtain" is a simply constructed remotely operated neutron shutter, which permits turning "on" and "off" the thermal-neutron streams to the thermal column and the experimental shielding facility.

Neutron curtain in the MTR. A sheet 6 ft 6 in. square and $\frac{1}{4}$ in. thick of boral mounted in an aluminum frame serves as the neutron curtain. During operation, this curtain is used, if desired, to cover the face of the thermal column and reduce the number of thermal neutrons entering the column. It has rollers at its four corners that travel in vertical guides mounted in the 3-in. space between the two 4-in. thermal-shield plates.

A vertical slot 2 in. by 7 ft in the biological shield, from the bottom of the thermal shield to the subpile room ceiling, houses the curtain when it is not in use. To shield against radiation streaming, the lower portion of this slot is filled with removable barytes concrete blocks 2 ft deep, supported from the subpile room ceiling by steel bars 6 in. square which serve as additional shielding.

A lifting mechanism mounted on the top of the reactor structure raises or lowers the curtain as desired. After the concrete and steel shielding at the bottom of the curtain slot is removed, the curtain can be lowered into the canal and a new curtain installed if necessary.

Provisions have also been made for installation of a curtain on the west side of the thermal shield at the face of the shielding facility; however, no curtain or lifting mechanism was furnished initially.

The neutron window. The neutron window is a sheet of lead 3 in. thick, placed between the permanent graphite and the thermal column to absorb gamma radiation but permit easy neutron passage. The lead sheet is mounted in a steel frame and placed in the (approximately) 6- by 6-ft hole in the outer thermal shield.

3-9 The Biological Shield. The biological shield (see Figs. 3-3 and 3-4) around the reactor reduces by absorption the high neutron and gamma fluxes escaping through the reflector. The shield is designed to reduce these radiations to a safe level for personnel and to a level at which sensitive instruments can be operated outside the shield.

The shield is penetrated by about 100 holes of various sizes. All the permanent liners for the experimental holes are embedded in the concrete and welded to the steel plates, which act as inner and outer form plates for the concrete. One of the major problems in the construction of the reactor was the alignment of these liners. As indicated previously, the top of tank section *E* was the only reference point available, and accuracy was essential if the HB and DB inner liners were to fit into their respective locations in the tank section *D* wall.

The MTR shield was built of barytes concrete in which the gravel part of the mix is approxi-

mately 93 per cent BaSO_4 . At first it was thought that only certain critical sections of the shield would be made of barytes, but owing to difficulty of mixed pours it was decided that a more satisfactory shield would be obtained if barytes was used throughout. In order to simplify the construction procedure and obtain a uniformly dense concrete, it was decided to use the "prepack" method for setting all the biological shield.

Experimental Facilities

To fulfill the purpose of the MTR, the experimental facilities are as varied and flexible as possible. These may be divided into several groups which are discussed briefly in the following paragraphs. They are described more fully in Part 6, Engineering Testing.

3-10 Reflector and Lattice Positions. In the reflector region inside the reactor tank certain sections are designed for easy removal. Materials can be irradiated by inserting them in these posi-

tions. To do this, special reflector pieces are made up of beryllium or aluminum with the proper size holes to handle the test equipment. Irradiations in these positions are generally carried out in aluminum cylinders or rings in which the material under investigation is encased. Similarly, special pieces can be inserted in lattice positions not occupied by fuel elements or shim rods.

Generally, experiments in these positions are of the prepost type, i.e., they are inspected before and after irradiation and do not require any communication during irradiation. It is possible, and it is being done, to bring a few small tubes out through the tank access holes so that thermocouple leads, electrical leads, and very small coolant pipes can be attached to the experimental material.

3-11 Horizontal and Down-beam Experimental Holes. There are 17 large experimental holes that lead from the reactor faces either to the reactor-tank wall or to the active lattice. These holes are described separately from the vertical

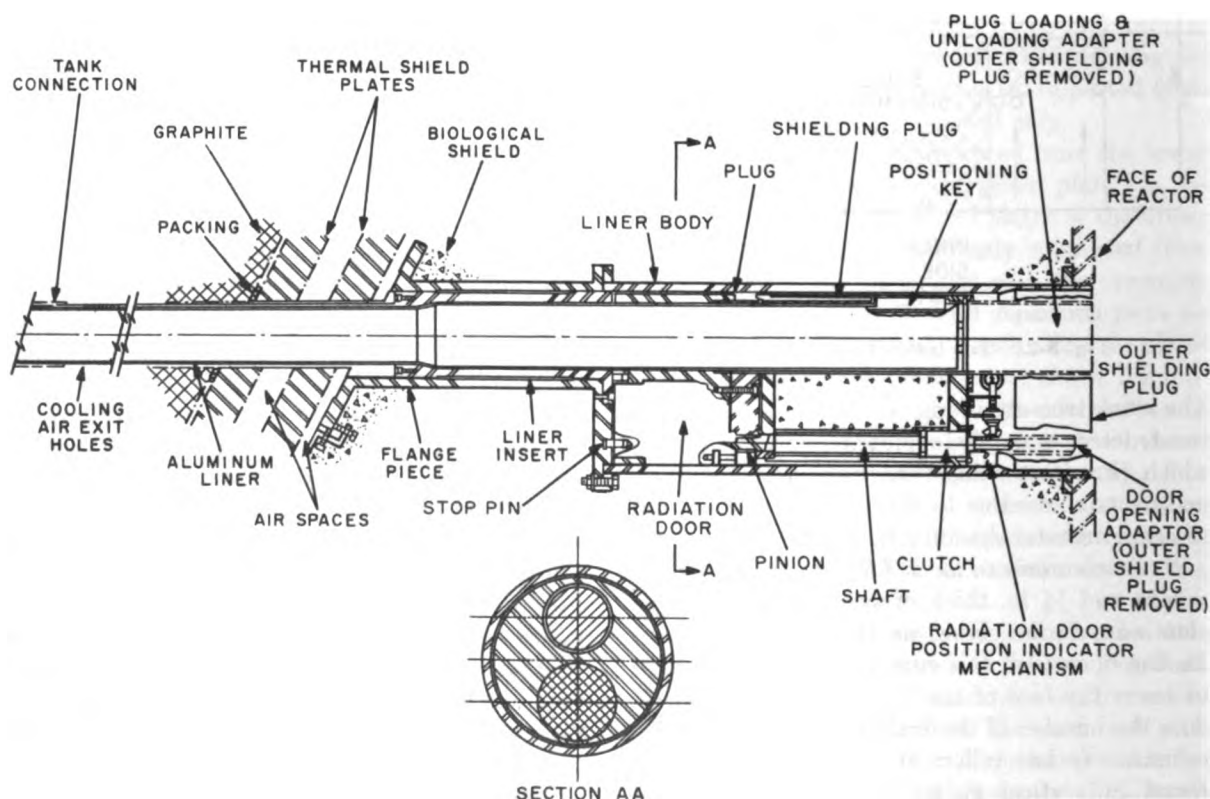


Fig. 3-22 Typical beam-hole liner and components.

Table 3-3 MTR Experimental Facilities
(Center line of reactor defined as elevation 100 ft 0 in.)

Facility designation	Size	Penetration	General description
HB (horizontal beam holes):			
HB-1.....	6 in. ID	To active section edge on reactor horizontal center line	Horizontal approach on line 27° north of east, intersecting edge of active section 10¼ in. north of reactor center line at elevation 100 ft 0 in.
HB-2.....	6 in. ID (8 in. to tank)	8 in. ID to tank wall, 6 in. ID to active section edge on reactor horizontal center line	Horizontal approach from east on east-west, center line of reactor at elevation 100 ft 0 in.
HB-3.....	6 in. ID	To active section edge on reactor horizontal center line	Horizontal approach on line 27° south of east, intersecting edge of active section 10¼ in. south of reactor center line at elevation 100 ft 0 in.
HB-4.....	6 in. ID	To active section edge on reactor horizontal center line	Horizontal approach on line 27° north of west, intersecting edge of active section 10¼ in. north of reactor center line at elevation 100 ft 0 in.
HB-5.....	6 in. ID (8 in. to tank)	8 in. ID to tank wall, 6 in. ID to active section edge on reactor horizontal center line	Horizontal approach from west on east-west, center line of reactor at elevation 100 ft 0 in.
HB-6.....	6 in. ID	To active section edge on reactor horizontal center line	Horizontal approach on line 27° south of west, intersecting edge of active section 10¼ in. south of reactor center line at elevation 100 ft 0 in.
DB (down-beam hole):			Located above center of reactor and in same vertical plane as corresponding HB facility. Inclines downward moving toward reactor and has a center line in a plane making an angle of 30° with horizontal and east-west center line of reactor. Extends inward through graphite to tank wall; a sealed can inside the tank goes from this point to intersection with the active section at elevation 100 ft. 9¼ in. Approximately 1¼ in. of aluminum and 20 in. of air lie between active section and innermost point of accessibility
DB-1 } DB-2 } DB-3 } DB-4 } DB-5 } DB-6 }	6 in. ID	Through graphite to reactor tank wall with a 2 in. ID can inside tank from tank wall to active section edge	<div style="display: flex; align-items: center;"> <div style="margin-right: 10px;"> Approach from NE Approach from east Approach from SE Approach from NW Approach from west Approach from SW </div> </div>
VH (vertical hydraulic rabbit holes):			
VH-1.....	1 in. ID	Reflector	Vertical facility, running height of reflector at point 16.707 in. south, 6.070 in. west of reactor center line. Specimens charged in basement for delivery by water to exposure point. Specimen cooled by water discharging past capsule container into reactor tank just above reflector. Specimens returned by reactor water pressure
VH-2.....	1 in. ID	Reflector	Same as VH-1, located 16.707 in. south, 6.070 in. east of reactor center line
VH-3 and VH-4.....	Approx. 1¼ in. diam	Reflector	Vertical hydraulics, similar to VH-1 and VH-2, occupying spare regulating rod spaces in reflector. Located 5¼ in. north or south and 9½ in. east or west of reactor center lines
HR (horizontal pneumatic rabbit holes):			
HR-1.....	1 in. ID	Reflector	East-west horizontal pneumatic tube through reactor structure and tank 1 ft 3½ in. south of reactor center line. Enters structure at elevation 99 ft 5½ in. and makes a 4-ft-radius Z bend to pass through tank and graphite at elevation 99 ft 2½ in.
HR-2.....	1 in. ID	Reflector	East-west horizontal pneumatic tube through reactor structure and tank 1 ft 3½ in. south of reactor center line. Enters structure at elevation 98 ft 9½ in. and makes a 4-ft-radius Z bend to pass through tank at elevation 99 ft 0½ in.
HR-3.....	4 in. ID	To reactor tank	Horizontal holes for oriented samples, approaching reactor radially from north face of structure on line 30° west of north at elevation 100 ft 0 in. extending inward to reactor tank
HR-4.....	4 in. ID	To reactor tank	Horizontal holes for oriented samples, approaching reactor radially from north face of structure on line 30° east of north at elevation 100 ft 0 in. extending inward to reactor tank
HT (horizontal through holes):			
HT-1.....	4¼ in. square inside	Reflector	Through hole from side to side of reactor structure, through reactor tank adjacent to north end of active section 1 ft 5½ in. from reactor center line at elevation 99 ft 3¼ in. (Dimensions to center line)

Table 3-3 MTR Experimental Facilities (Continued)

(Center line of reactor defined as elevation 100 ft 0 in.)

Facility designation	Size	Penetration	General description
HI (horizontal instrument holes):			
HI-2.....	10½ in. ID	Graphite	Horizontal through reactor structure, running north-south, 4 ft 6 in. east of pile center line at elevation 104 ft 3 in.
HI-3.....	10½ in. ID	Graphite	Horizontal through reactor structure, running north-south, 4 ft 6 in. west of pile center line at elevation 104 ft 3 in.
HG (horizontal graphite holes):			
HG-1.....	8 in. ID	Graphite	Through reactor structure, running east-west, 5 ft 3 in. north of reactor center line at elevation 101 ft 6 in.
HG-2.....	8 in. ID	Graphite	Through reactor structure, running east-west 5 ft 3 in. north of reactor center line at elevation 98 ft 6 in.
HG-3.....	4 in. ID	Graphite	Through reactor structure, running east-west 5 ft 1½ in. south of reactor center line at elevation 101 ft 2 in.
HG-4.....	4 in. ID	Graphite	Through reactor structure, running east-west 5 ft 1½ in. south of reactor center line at elevation 98 ft 10 in.
HG-5.....	6 in. ID	Graphite	Horizontal, approaching reactor radially from south face of structure on line 30° east of south at elevation 100 ft 0 in. extending inward to reactor tank
HG-6.....	6 in. ID	Graphite	Horizontal, approaching reactor radially from south face of structure on line 30° west of south at elevation 100 ft 0 in. extending inward to reactor tank
HG-9.....	10 × 16 in.	Through graphite to tank wall. 3 sealed cans inside tank 5¼ by 12 in. from tank wall to edge of active section. Rectangular, long axis is vertical	Horizontal hole to reactor tank, 1½ in. east of reactor center line and at elevation 99 ft 9¼ in. The center line of the facility is parallel to the north-south center line of the reactor. There are approximately 3 in. aluminum and 10 in. of air between active section and innermost point of accessibility. Hole located on south side of reactor
GT (gamma thimble):			
GT-1.....	6 in. ID	Discharge water line	Vertical thimble from top of structure in north-west discharge water line for gamma exposures. Located 7 ft north of east-west center line and 8 ft 6 in. west of north-south center line. Extends to elevation 109 ft 3 in.
GT-2.....	6 in. ID	Discharge water line	Vertical thimble in south-east discharge water line. Located 7 ft south of east-west center line and 8 ft 6 in. east of north-south center line. Extends to elevation 109 ft 3 in.
T2H (thermal column horizontal):			
T-2.....	6 ft square	To thermal shield	6-ft-square inner end, expanding to approximately 6 ft 4 in. at outer end, south side of reactor on north-south center line
T2H-1.....	4 in. square	T-2	Horizontal stringer through reactor structure, running east-west, 13 ft south of reactor center line at elevation 101 ft 4 in.
T2H-2.....	4 in. square	T-2	Horizontal stringer through reactor structure, running east-west, 13 ft south of reactor center line at elevation 98 ft 8 in.
T2H-3.....	4 in. square	To thermal shield	Through thermal column horizontally to thermal shield. 1 ft 8 in. west of reactor center line at elevation 101 ft 10 in.
T2H-4.....	4 in. square	To thermal shield	Through thermal column horizontally to thermal shield. 1 ft 11 in. east of reactor center line at elevation 101 ft 10 in.
T2H-5.....	4 in. square	To thermal shield	Through thermal column horizontally to thermal shield. 1 ft 8 in. west of reactor center line at elevation 100 ft 0 in.
T2H-6.....	4 in. square	To thermal shield	Through thermal column horizontally to thermal shield. 1 ft 11 in. east of reactor center line at elevation 100 ft 0 in.
T2H-7.....	4 in. square	To thermal shield	Through thermal column horizontally to thermal shield. 1 ft 8 in. west of reactor center line at elevation 98 ft 2 in.
T2H-8.....	4 in. square	To thermal shield	Through thermal column horizontally to thermal shield. 1 ft 11 in. east of reactor center line at elevation 98 ft 2 in.
T2V (thermal column vertical):			
T2V-1.....	1 ft square	T-2	Vertical stringer, extending downward through thermal column from top of structure, located 10 ft 3 in. south of reactor center line, and 1 ft 1½ in. east of reactor center line
T2V-2.....	1 ft square	T-2	Vertical stringer, extending downward through thermal column from top of structure, located 10 ft 3 in. south of reactor center line, and 10½ in. west of reactor center line
VN (vertical instrument holes):			
VN-1 } VN-2 } VN-3 } VN-4 } VN-5 } VN-6 }	3 in. ID	Graphite pebble zone	Straight slanting holes, the center lines of which form the frustum of a cone, 8 ft 0 in. in diameter at the top of the reactor structure and 5 ft 4 in. in diameter at elevation 100 ft 5 in. where the holes terminate. Center line of frustum is coincident with reactor center line.

Table 3-3 MTR Experimental Facilities (Continued)
(Center line of reactor defined as elevation 100 ft 0 in.)

Facility designation	Size	Penetration	General description
VG (vertical graphite holes):			
VG-1.....	4 in. ID	Graphite pebble zone N-E side Graphite pebble zone S-E side Graphite pebble zone S-W side Graphite pebble zone N-W side Graphite N-W side	Vertical straight hole from top of reactor structure through pebble zone to bottom of graphite reflector. 3 ft 1½ in. from north-south and east-west center lines
VG-3.....	4 in. ID		
VG-5.....	4 in. ID		
VG-6.....	4 in. ID		
VG-2.....	4 in. ID		
VG-4.....	4 in. ID	Graphite S-E side	Vertical through hole from top of reactor structure into canal, 4 ft 2 in. from east-west center line, 1 ft 4 in. from north-south center line
VG-7, 8, 10 to 20....	2½ in. ID	Graphite pebble zone	Vertical through hole from top of reactor structure into canal, 4 ft 2 in. from east-west center line, 1 ft 6 in. from north-south center line
VG-9.....	10 in. square	Graphite to HG-9	Straight slanting holes, the center lines of which form a frustum of a cone with a vertical axis coincident with the reactor-tank center line. The holes commence at the top of the structure, the frustum at this point being 8 ft 0 in. in diameter, and terminate at elevation 98 ft 4 in. where the frustum is 5 ft 8 in. in diameter
VG-23 to VG-62.....	2 in. ID	Graphite	Vertical access to HG-9 hole located 4 ft 5 in. south of east-west center line and 1½ in. east of north-south center line
VC (vertical graphite holes):			Vertical stringers from top of structure through graphite.
VC-1 to VC-13.....	2 in. ID	Concrete	Vertical exploratory holes from top of structure through concrete to elevation 92 ft 0 in.
SF (shielding facility) . .	6 ft square	To thermal shield	Horizontal facility, north side of reactor on north-south center line
VT-2 (vertical through holes).....	2½ in. ID	Through active section	Vertical through hole occupying any spare shim-rod space in active section

graphite holes because their size and locations require special design, shielding, and handling techniques. They are characterized by the presence in each of a special radiation door, the sole purpose of which is to minimize danger to personnel in the Reactor Building during plug-handling operations. Figure 3-22 shows a typical layout of a horizontal beam hole. Because these holes provide the largest volumes of high flux in the reactor, they are the most dangerous, and their use will probably be correspondingly difficult and complicated. They are as shown in Table 3-3.

1. Six horizontal beam holes HB-1 to HB-6. These holes extend horizontally from the reactor face to the active lattice. Holes HB-2 and HB-5 are 8 in. in diameter to the reactor-tank wall and 6 in. in diameter from the reactor-tank wall to the active lattice. Holes HB-1, HB-3, HB-4, and HB-6 are 6 in. in diameter throughout.

2. Six down-beam holes DB-1 to DB-6. These are inclined holes 6 in. in diameter which extend from the upper north and south faces to the reactor tank. They are located directly above the horizontal beam holes. Inside the tank wall they

are essentially extended to the lattice by sealed empty aluminum cans, 2 in. ID, in the beryllium reflector.

3. Two horizontal graphite holes HG-5 and HG-6. These are horizontal holes 6 in. in diameter which extend to the reactor tank.

4. Two horizontal rabbit holes HR-3 and HR-4. These are horizontal holes 4 in. in diameter which extend to the reactor tank.

5. One horizontal through hole HT-1. This is a hole $4\frac{11}{16}$ in. square which extends from one reactor face to the opposite face. It goes through the reactor tank and is adjacent to one side of the active lattice.

In formulating the design of these holes, considerable thought was given to the future needs of the experimenters who will use these facilities. Other factors that had a strong influence on the experimental-hole design are radiation hazards, material selection, cooling requirements, and the configuration of the reactor itself.

In addition to these main experimental holes there are six more horizontal through holes. Four of these, HG-1 to HG-4, are for experimental use;

the other two, HI-2 and HI-3, are for reactor-control instruments. HG-1 and HG-2 go through the full length of the east graphite wall, and HG-3 and HG-4 penetrate the west wall. The first two holes are 8 in. in diameter; the latter two are 4 in. in diameter through the graphite.

Access to HI-2 and HI-3 is obtained through the east and west faces of the reactor structure. These holes are used for the compensated ionization chambers for the log N and galvanometer circuits of the reactor-control system. Both holes have a minimum diameter of $10\frac{3}{4}$ in. and are plugged only through the biological and thermal shields. The plugs used are barytes concrete tipped with an aluminum casting 30 in. long. A hole 4 in. in diameter runs through the center of the plug to within 4 in. of the inner end. This allows the tip of the ionization chamber to be inserted to the inner side of the thermal shield. An adjustment of the ionization-chamber insertion rod enables the chamber to be moved back as much as 18 in. from the inner position.

3-12 Vertical Experimental and Instrument Holes. The MTR has 71 experimental and instrument holes accessible from the top of the reactor. Some of these penetrate into the pebble zone and some into the permanent graphite; two are also accessible from the subreactor room. The dimensions range generally from 2 to 4 in. in diameter. Four of these holes are 6-in. thimbles extending into the exit-water lines, two for measuring exit-water activity and two to provide high-gamma-flux experimental facilities. One hole, 12 in. square, penetrates to the inside tip of the HG-9 hole coming through the thermal column. Two other holes, 12 by 12 in., also penetrate vertically downward into the thermal column.

3-13 Thermal Column. The thermal column is 6 ft square and 8 ft long outside a lead window 4 in. thick. The outer face is protected by a lead shielding door 10 in. thick. Six graphite stringers, 4 by 4 in., penetrate the thermal column to the inner lead window. A hole, HG-9, 9 by 15 in., penetrates the column to the reactor-tank wall. Two holes, 4 by 4 in., at 90° to the column penetrate from reactor face to reactor face. In addition, the two 12- by 12-in. vertical holes mentioned previously are available.

At one time there was a second thermal column in the design. Later it was converted to a facility for testing shielding materials. Subsequently most of the features of this facility were canceled. At present the 6-ft-square hole is filled with barytes concrete blocks. It can readily be converted to another thermal column.

3-14 Shuttle Tubes. Two pneumatic shuttle (rabbit) tubes are provided, HR-1 and HR-2. These are horizontal tubes of 1 in. ID which extend completely through the structure and pass within 1 in. of the active lattice. These are in addition to HR-3 and HR-4, which approach the tank wall radially and have been mentioned earlier. Actually, any beam hole can be adapted to provide shuttle facilities. The pneumatic shuttle tubes are generally thought to be useful for very short-time irradiations.

Hydraulically operated shuttles are also provided. These are for irradiations of small samples which require times up to the order of 2 weeks and require removal between normal reactor shutdowns. They are two vertical tubes of 1 in. ID running through the bottom plug into the beryllium reflector adjacent to the active lattice. In addition, tubes 1.31 in. in diameter can be used in any of the four regulating-rod positions which are not in use. These hydraulic shuttles are accessible through a short extension of the main canal in the reactor-building basement.

Reactor Cooling

3-15 Process-water-system Description. The process-water system is the primary cooling system of the materials-testing reactor. The system provides a flow of water through the active lattice and other parts of the reactor that require water cooling. The schematic diagram of the MTR process-water system presents an over-all picture of all the vessels, pumps, pipelines, and auxiliary equipment involved in the MTR process-water system (Fig. 3-23).

Water at 100°F is pumped up to the working reservoir from the 60,000-gal sump tank by means of two horizontal centrifugal pumps, each with a capacity of 10,000 gal/min and a head of 225 ft. A third pump is available in the event that either of the operating pumps fails or is shut down for

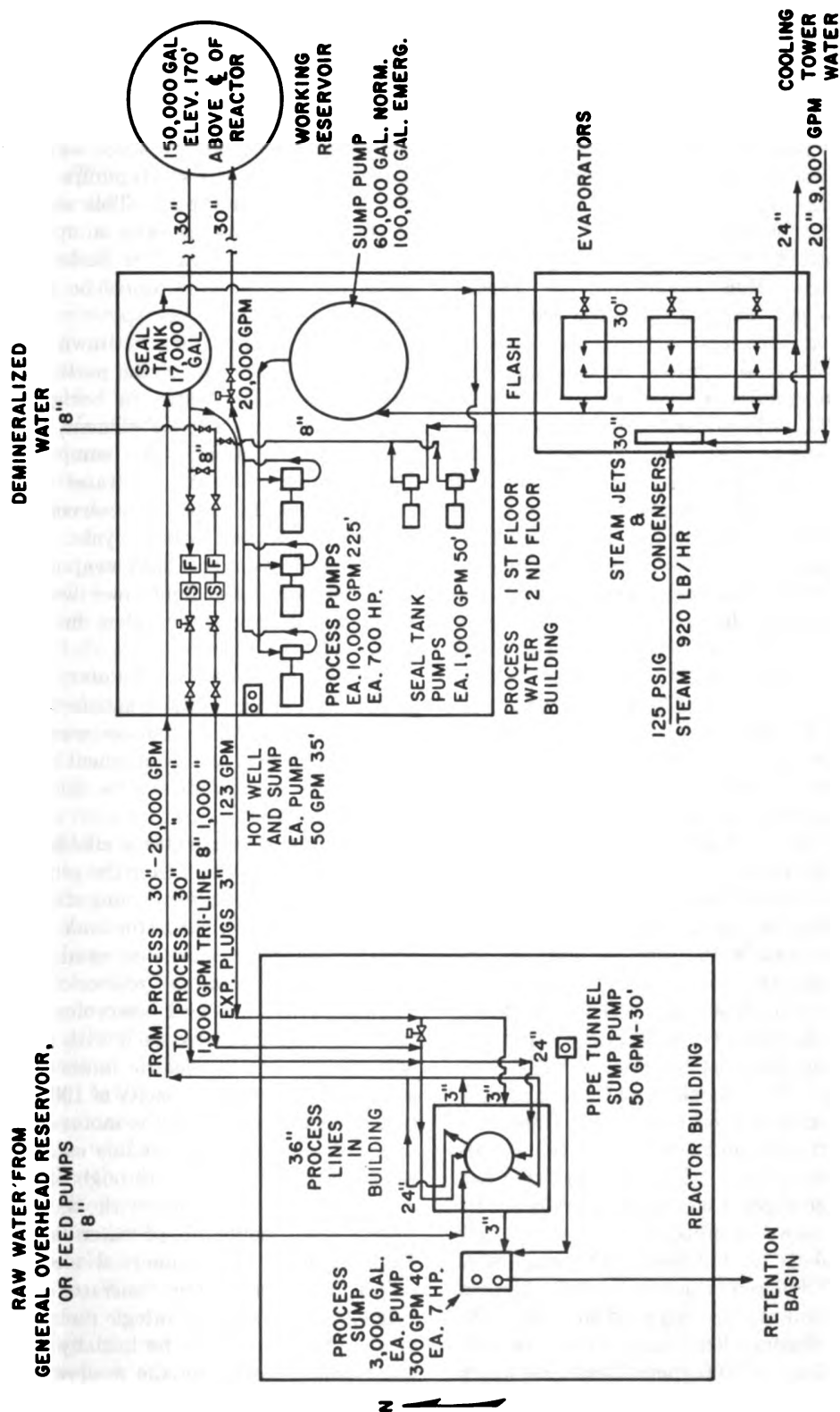


Fig. 3-23 Schematic diagram of process-water system.

maintenance. The water from the 150,000-gal working reservoir flows, under a normal head of 170 ft, directly to the reactor tank through a 30-in. stainless-steel pipe outside the Reactor Building and a 36-in. stainless-steel pipe beneath the Reactor Building. A flow instrument, a strainer, and a flow-control valve operated by an electric motor are located between the working reservoir and the reactor. These three items are blocked off in a pit by gate valves so that the water may be drained from them and they may be inspected or repaired. The 36-in. process-water line is split into two 24-in. stainless-steel lines below the basement floor. These lines go up through diagonally opposite corners of the reactor biological shield and connect to the top section of the reactor tank. The water descends through the reactor-tank sections, the active lattice, and the beryllium reflector and emerges from the bottom of the reactor tank through two 24-in. lines. The two exit lines are joined to a 36-in. line at the same location as that where the inlet line diverges from a 36-in. line into two 24-in. lines. The exit 36-in. line carries the water to the 17,000-gal seal tank in the Process-water Building.

The pressure losses between the working reservoir and the exit of the reactor tank are due almost entirely to the velocity head and friction losses across the active and beryllium sections, with relatively small losses caused by the conveying pipeline. There are block valves and a flow-control valve in the inlet line to the reactor, but there are no obstructions in the pipeline from the reactor to the seal tank.

There is an 8-in. line connected to the 30-in. inlet line at a point upstream from the valve pits and to the 36-in. line near the reactor. This 8-in. line provides a flow parallel to the main flow stream and contains two block valves, a flow instrument, a strainer, and a control valve. It can carry 1000 gpm and is used in the event that the main water-line valve is closed off, thereby assuring a continuous water supply to the reactor at all times, particularly at shutdown. This same line, which allows 1000 gpm of process water to bypass the main stream, can also be used to carry 1000 gpm of fresh demineralized water during periods of reactor flushing or 1000 gpm of seal-tank water for recirculation through the seal tank and the reactor only.

The pipe tunnel beneath the Reactor Building is about 55 ft below the center line of the active section and lies on bedrock. At the low point in the pipe tunnel there is a single sump pump with a capacity of 50 gpm which pumps out any water that leaks into the tunnel. This water is pumped to the 3000-gal process-water sump, which is also located beneath the Reactor Building. The two 36-in. lines in the pipe tunnel lie one above the other; the inlet line is on top.

Water in the seal tank is drawn into the flash evaporators by means of the partial vacuum existing there. The water, in being flashed at a pressure that yields 100°F effluent, falls from the flash evaporators into the sump tank below. From the sump tank the water is once more pumped up to the working reservoir to complete the process-water cooling cycle. The reactor heat, liberated in the flash evaporators, is carried away by the cooling-tower water circulated through the condenser tubes in the evaporators.

Stand-by and emergency features. Certain provisions are made to ensure satisfactory operation of the process-water system under emergency conditions. The primary requirement in emergencies is that there must always be some water flow through the reactor.

In addition to the water available in the working reservoir, raw water from the general overhead reservoir is available for cooling the reactor by a direct 8-in. line to the reactor tank. In the event both overhead tanks are destroyed, water can be pumped directly to the reactor from the two 500,000-gal ground-level reservoirs by means of two centrifugal pumps, each with a capacity of 850 gpm, driven by electric motors, or one centrifugal pump with a capacity of 1000 gpm, driven by a gasoline engine. If the motor-driven pumps in the demineralized-water line can be used, the reactor can be supplied through the two 24-in. inlet process-water lines with water from the 100,000-gal demineralized-water storage tank. Within 30 sec of commercial electrical power failure, a diesel electric generator set will provide electric power to strategic pumps and equipment. The reactor can be initially cooled under these circumstances by the reserve water in the working reservoir and then by 1000 gpm of process water recirculated by one of the seal-tank pumps,

one of which is driven by an electric motor and the other by a gasoline engine.

The loops in the 24-in. water lines in the reactor structure help keep the active lattice under water at all times. This assures natural convection water cooling during an emergency shutdown and thereby prevents melting of the fuel assemblies. Should local water boiling occur, the localized steam pressure could conceivably force the water out of the active section and cause melting. However, sufficient pressurized water sources are available to ensure practically some water flow through the active lattice. A 2-in. line between the top of the reactor tank and the water loop is provided to prevent siphoning of the water from the active lattice if the process-water pipeline should be severed. In the event of any major emergency due to electrical failure, steam failure, etc., the reactor power will be nulled and shutdown cooling will be the only requirement.

Instrumentation and flow balance. To maintain a continually balanced water flow through the process-water system, a scheme was established in which water is pumped from the sump tank to the working reservoir. The valve that determines the amount of water lifted is governed by the level of water in the working reservoir. The water-flow rate from the working reservoir to the reactor at a given control-valve setting and a constant head is uniform. The water flows from the reactor to the seal tank and then through the flash evaporators into the sump tank. Any overflow from the seal tank also goes to the sump tank. Any shortage of water in the seal tank is automatically compensated for by a decreased flow to the flash evaporators, because the flash evaporators are fed water at a rate that is determined by the level of water in the seal tank and the pressure in the flash evaporators. In this manner the water flow through the whole system is kept in equilibrium. Normally the sump tank contains 60,000 gal of water. However, to accommodate undue surges and accumulations of water during emergencies such as electrical failure, the capacity of the sump tank is 100,000 gal.

Instrumentation on the flash evaporators provides the maximum possible vacuum. The flow of water to the flash evaporators is adjusted by means of valves in the inlet lines. The temperature of the water leaving the flash evaporators

determines, by means of a thermal control, the quantity of cooling water passing through the condensing tubes.

The following instrumentation is related to the working reservoir. A temperature-indicating controller limits the lowest water temperature in the working reservoir by allowing steam to flow into a heating coil at the base of the reservoir. The floating roof of the working reservoir is equipped so that at water levels exceeding normal a mechanical switch shuts off the main process-water lift pumps. Within the normal operating range, the level in the working reservoir is maintained by an instrument which governs the opening of the main valve controlling the water flow up into the working reservoir. At lower water levels in the working reservoir the reactor is abruptly shut down and the main water valve is closed in 30 sec. The water flowing to the reactor passes through a pit in which there are two block gate valves which are pneumatically operated. The purpose of these block valves is to isolate the pit so that the portion of pipe between the block valves can be drained and inspected. Between the block valves, going downstream, there are, in order, a flow recorder, a motor-operated control valve, and a strainer. Adjacent to the pit are taps for instruments which make continuous recordings of the water resistivity and pH. Upstream of the first block valve is a 3-in. tap for water used in cooling the experimental plugs; the full head due to the height of water in the working reservoir is always available at this tap. The only valves that control the flow in this line are at the experimental plugs. An 8-in. bypass line before the main motorized valve allows a flow of about 1000 gpm of water to the reactor when the main valve is shut. This line joins another 8-in. line from the demineralized-water supply and an 8-in. line from the seal-tank pumps. Each of these three lines has an on-off remotely controlled pneumatic valve. Immediately after shutdown, process water goes through the 8-in. line directly to the reactor. Shortly thereafter, demineralized water is used to purge the reactor-seal-tank system. This 8-in. line enters the reactor-cooling-water line at the point where the two 24-in. lines branch from the main 36-in. water-inlet line. It also passes through the valve pit with the main water line and has two block gate valves. Going downstream there are,

in order, a flow meter, an 8-in. pneumatic flow-control valve, and a strainer.

The temperature and pressure of the water entering and leaving the reactor are recorded. The signal of the temperature difference between entering and leaving reactor water and that of the water flow are multiplied in a power recorder. The power of the reactor is also obtained from instruments measuring the gamma intensity of the water leaving the reactor and the neutron flux. Instruments also record the temperature rise of the water in going through each fuel assembly, record the radiation intensity of the water emerging from each fuel assembly, and indicate the amount of flow through each assembly.

A flow recorder is provided on the water lines from each experimental plug, and unless the flow through the plugs is satisfactory, the reactor cannot be started. A water meter in the line supplying demineralized water to the rabbits and experimental plugs keeps an inventory of the amount of water introduced into the process-water system in this manner.

The water leaving the reactor enters the seal tank in the Process-water Building. In the seal-tank entrance there is a radiation recorder for detecting fission breaks. The seal tank has an inlet line and meter for supplying and metering make-up water. It also has a temperature recorder and a level-indicating alarm. All pumps in the process-water system have pressure gauges. The flow to the flash evaporators from the seal tank is controlled by a valve and is indicated by an instrument in the outlet line from the flash evaporators. The flash evaporators are operated at full vacuum potential. Instruments indicate the vacuum on each evaporator and control the amount of cooling water going to each evaporator condenser. The pressure of the steam going to the vacuum system is recorded and controlled. Temperatures in various points of the inter- and after-condenser vacuum-cooling system are indicated. The vapors discharged from the vacuum system are monitored by a radiation instrument which detects radiations and gives an alarm if large amounts of activity are discharged from the Process-water Building into the atmosphere. The sump tank has a level-indicating control alarm and a temperature recorder. A flow recorder is provided for the purge from the sump tank.

Water-cooled experimental plugs and rabbits. In the reactor there are seven water-cooled experimental plugs which require a total of about 125 gpm of cooling water. Water to cool these plugs is tapped from the main process-water line upstream of the valve pit. This assures that there will always be pressure available sufficient to force the water through the plugs, even during shut-down periods. The water leaving the experimental plugs flows into the process-water exit line from the reactor. In case it is not advisable to return the plug-cooling water to the process-water system, the water can be purged directly to the process-water sump. Demineralized water, which is one of the service facilities provided at beam holes, may also be used for plug cooling. During electrical failures these experimental plugs must have cooling water, and because demineralized-water pressure will fail at this time, a bypass valve is provided from the process-water system to the demineralized-water system so as to maintain cooling of the plugs.

Process-water Building. The 20,000-gpm flow of MTR process water passes through the Process-water Building before entering and after leaving the reactor. This building contains the equipment for controlling the water flow, the pumps for lifting the water to the working reservoir, the shut-down and emergency cooling-water pumps, and the flash evaporators for cooling the process water. Figures 3-24 and 3-25 show Process-water Building floor plans and building sections, respectively.

The seal tank receiving the water from the reactor section and the sump tank receiving the process water from the flash evaporators are enclosed in this building for shielding and instrument-maintenance reasons. The equipment is located on seven main levels so as to obtain building compactness and to achieve proper hydrostatic balance between the seal-tank level, suction lift to the flash evaporators, and barometric discharge to the sump tank.

The 30-in. water line from the Reactor Building enters the Process-water Building through the control-valve pit and then discharges into the 17,000-gal seal tank. From this tank the process water flows successively through the flash evaporators, sump tank, process-water pumps, and working reservoir, and then to the reactor through a control valve in the Process-water Building.

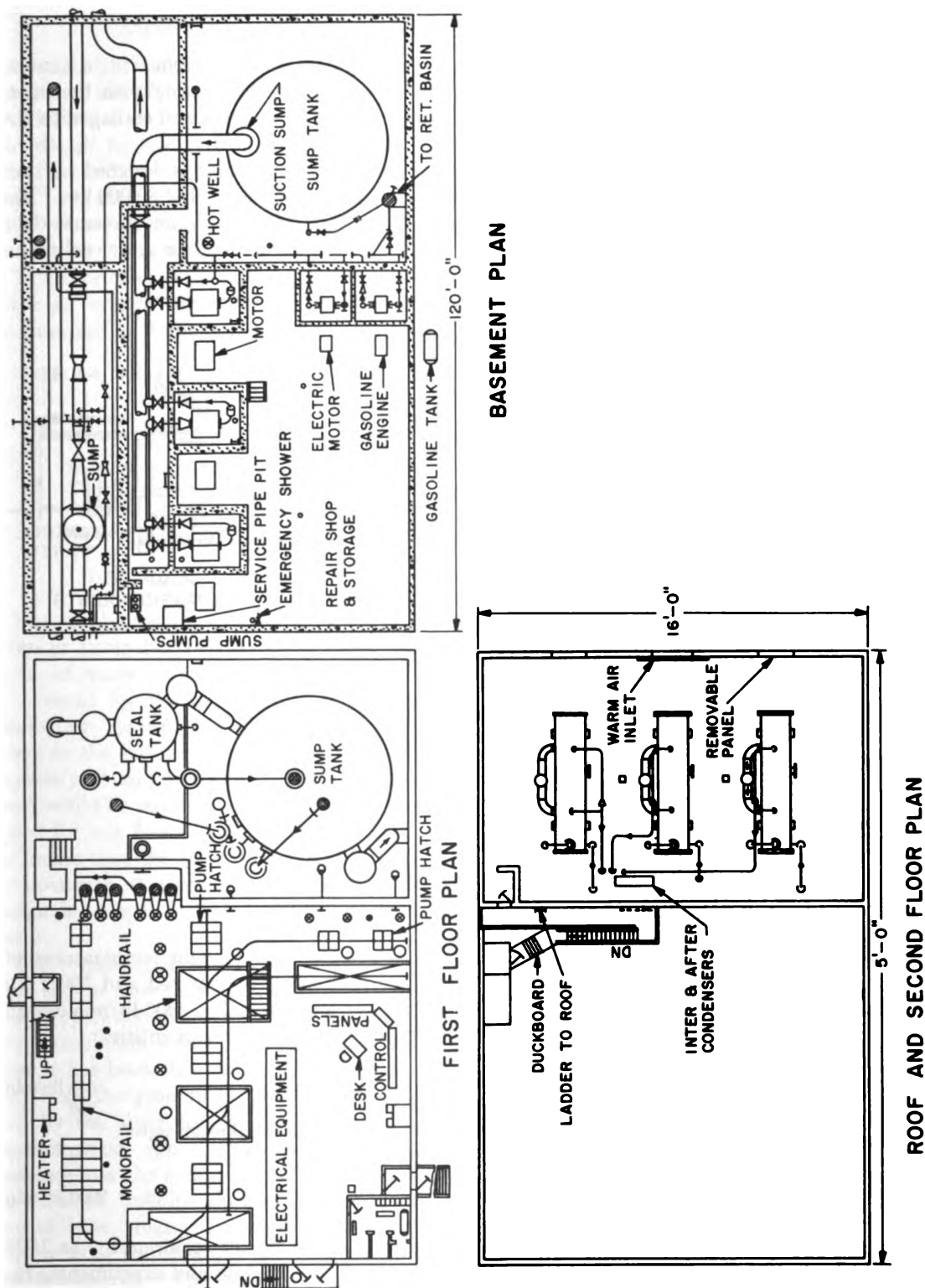


Fig. 3-24 Floor plans for Process-water Building.

All piping and pumps are shielded with about 1 ft of concrete, and the valve-control handles are extended through the shielding. All instrumentation and controls for maintaining proper process-water temperatures and flows are located centrally in the Process-water Building instrument room. Overhead cranes and removable floor slabs provide accessibility to shielded pipes, valves,

The description of the system will in general follow the air-flow path, starting with the inlet-air filters at the reactor faces and ending with the stack.

Air-flow data. The reactor is designed for normal operation at a power level of 30,000 kw. The design air flow of 2000 lb/min and pressure drop of 55 in. of water through the reactor cooling-air

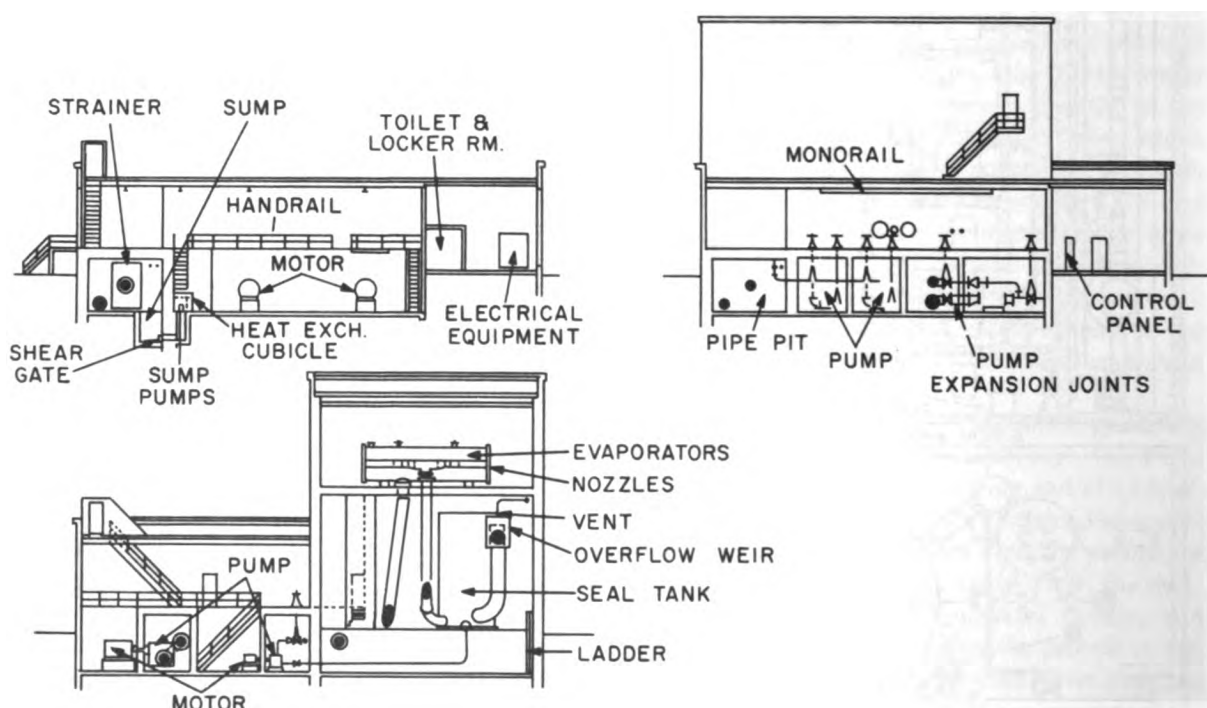


Fig. 3-25 Sectional views of Process-water Building.

pumps, etc. Each pump cubicle is shielded from its neighboring cubicle so that repairs can be made during reactor operation.

3-16 Reactor Cooling-air System. Introduction.

Immediately surrounding the reactor tank is the graphite reflector. This reflector is composed of a removable bed of graphite balls 1 in. in diameter adjacent to the tank and a section of graphite blocks between the balls and the thermal shield. Heating due to the absorption of gamma radiation makes it necessary to cool the graphite reflector. Because of the relatively low heat generation, air cooling is satisfactory. This section will deal with the flow path and processing of the air going through the reactor cooling-air system.

system are based on maximum temperatures of 570°F in the graphite pebble bed and 500°F in the permanent graphite. The 2000-lb/min design flow consists of the following quantities:

Air flow through graphite.....	1610 lb/min
Air to cool top thermal shield.....	250
Allowed for leakage inward through experimental holes, etc.....	140
Total.....	2000 lb/min

Calculations gave the air requirements as 1526 lb/min to cool the graphite and experimental facilities and 174 lb/min to cool the top thermal shield, for a total air flow of 1700 lb/min. Also, critical experiments indicated that the heat gen-

eration in the reflector would be somewhat less than that used for these calculations and the air requirements would therefore be less. However, the design figures were maintained at 2000 lb/min flow and 55 in. of water pressure drop since it may be desirable and possible to operate the reactor at a power level as high as 60,000 kw at some time in the future.

The approximate pressure drops through the reactor air system for a flow of 2000 lb/min are summarized in Table 3-4.

Table 3-4 Approximate Pressure Drops in Reactor Air System

Part of system	Pressure drop, in. H ₂ O
Inlet filters and ducts.....	1.6
Thermal shield.....	0.4
Graphite reflector.....	29
Exit ducts and stack.....	5.4
Total.....	36.4

The total pressure drop of 36.4 in. of water given in Table 3-4 is considerably lower than the 55 in. of water design value. Two factors enter to account for the difference. (1) The reactor cooling-air system was designed without any filters on the exit-air stream. If it should become necessary to add filters in the future, additional head would be required. The additional head allowed for exit filters is about 8 in. of water. (2) To assure that the air system would be adequate for operation at a power level of 60,000 kw, the design head was further increased to 55 in. of water.

The foregoing discussion has been concerned with the air flow through the reactor air system during operation of the reactor. It is also necessary to maintain a flow of air through the reactor after it has been shut down. This flow removes heat from the graphite reflector, which is generated by the absorption of gammas from fission-product decay and from induced activity, and also prevents the escape of radioactive dust into the Reactor Building during removal of experimental hole plugs. To cool the reactor adequately after shutdown, a flow of about 135 lb/min is required. The flow necessary to ensure that radioactive dust does not escape into the Reactor Building is based on air velocity of 150 fpm directed inward along the annulus between a

plug and liner. The air flowing into this annulus flows through holes at the inward ends of the liners and joins the reactor cooling air flowing upward through the graphite pebbles. A satisfactory design of the experimental-hole liners and plugs was obtained when the flow through the reactor was about 400 lb/min; therefore, this is the design flow through the reactor during shutdown. This is also the flow desired through the reactor in the event of an emergency shutdown of the reactor. The pressure drop through the graphite for a flow of 400 lb/min is about 0.5 in. of water, and the design pressure drop through the entire system is about 2 in. of water.

During reactor operation there will always be active particles within the air passages. To prevent these active particles from entering the Reactor Building, a basic requirement of the reactor air system was that there always be a flow of air in the proper direction. Upon failure of the main exhaust blowers and the auxiliary fans, this flow will be induced by whatever stack draft is available.

After reactor startup and operation for a short time (long enough so that the reactor, reactor air system, and stack are hot), there is a stack draft of about 0.5 in. of water available. This is based on air temperatures of 200°F in the stack and 90°F atmospheric. It is estimated that this draft would produce a flow of 2000 to 3000 cfm through the reactor in the event of failure of all blowers and fans. This flow is sufficient to meet shutdown cooling requirements. However, because of the possibility of a complete power failure within a short time after reactor startup, in which case there would be essentially no stack draft, and because of the problem of radioactive dust escaping into the Reactor Building, the reactor cooling-air system does not rely on stack draft to produce the necessary shutdown air flow. This flow is provided by either of the auxiliary fans. Only as a last resort, i.e., in the event of complete failure of the blowers and fans, is the available stack draft utilized.

Additional Data

Air temperatures:

Inlet to reactor.....	75–100°F
Outlet from reactor.....	185–210°F
Site elevation.....	4930 ft
Barometric pressure.....	12.25 psi

Inlet-air filters. Air used for cooling a reactor will emerge radioactive. The activity is due partially to gaseous atoms naturally occurring in air, such as argon, and partially to dust in the air. Not much can be done to reduce the activity from the first, but the latter can be reduced considerably by suitable filtering.

In the first considerations of the radioactivity problem of the MTR cooling-air system, a filter design was included in the exhaust side of the system. Further consideration of the problem along with economy measures prompted omission of the filters in the exhaust side of the reactor cooling-air system.

The only filters included in the reactor air system are the glass-wool filters at the inlets to the system. A brief description of the processing the air encounters before entering the reactor follows.

Normal atmospheric air enters the Reactor Building ventilation system through coarse fiberglass or wire-mesh prefilters. The air then goes through an electrostatic filter of 90 per cent discoloration efficiency before passing into the Reactor Building. The Reactor Building is pressurized to about $\frac{1}{2}$ in. of water so that leakage in through windows, doors, etc., will be kept to a minimum. The air entering the reactor is taken directly from the Reactor Building and is filtered at the inlets by glass-wool filters. There is an additional small amount of air (about 2 per cent) which enters the air system via experimental holes without passing through the glass-wool filters.

The elimination of the exhaust side filters, from the technical standpoint, is justified principally by the following:

1. The major portion of air entering the reactor has been filtered to some degree.

2. The air does not make direct contact with the fuel elements, and thus fission products from ruptured fuel elements will not enter the air stream.

3. The air flows in direct contact with the graphite reflector.

4. The air flows in direct contact with the thermal shield.

5. The air is susceptible to pollution from accidents occurring in the experimental facilities.

6. The level at which the effluent air is ejected into the atmosphere is 250 ft.

Should it be necessary to add exhaust filters at some later date, there is sufficient space for them adjacent to the Blower and Fan House.

The inlet filters are mounted in the inlets in the reactor faces. The filters are Airmat type PL-24 units made by American Air Filter Company. Six units are mounted in an opening 2 by 12 ft in each pile face, giving a total of twenty-four 2- by 2-ft units. The face velocity of the air entering the filters is about 350 fpm. The pressure drop through the clean filters is approximately 0.1 in. of water.

System within reactor structure. The system of duct work within the reactor is shown by Fig. 3-27, and the schematic flow of air within the reactor is shown by Fig. 3-26. In the design of the system within the reactor it was necessary to meet the following requirements:

1. The air must enter the pile structure at a level higher than 8 ft above the center line of the active lattice, and the exit ducts must loop upward to a level at least that high in order for it to be possible, in so far as the air system is concerned, to flood the interior of the thermal shield with water.

2. All ducts must be arranged within the shield so that radiation outward, both in straight-line paths and in "shine" through the ducts, will not be sufficient to bring radiation at any point outside the shield above permissible levels.

3. The air must enter the thermal shield around the outer edge of its top and pass downward, cooling the inner and outer walls and possibly the top and bottom of the thermal shield before passing into a plenum chamber under the graphite. The air then must flow upward through the graphite to a plenum chamber above the graphite and out through exit ducts.

4. Exit air ducts must be shielded from the outside by the equivalent of 1 ft of concrete.

5. Arrangement of the air ducts within the reactor structure must not interfere with any experimental facilities.

The design shown in Figs. 3-26 and 3-27 essentially meets all the foregoing requirements.

The inlet system consists of four sets of filters, plenum chambers, ducts and manifolds, one set in each wall of the reactor structure. The air enters each set through louvers located near the top

of each pile face. Back of each set of louvers are six filters each 2 ft square and a plenum chamber 2 by 2 by 12 ft. Each plenum chamber is joined to the top edge of the thermal shield by five or six ducts 8 in. square and a manifold which is continuous around the top of the thermal shield.

proper is shown in Fig. 3-26. The main air stream flows down between the walls of the thermal shield, and some then passes directly into a plenum chamber beneath the graphite while a portion flows between the top and bottom plates of the bottom thermal shield before passing into the

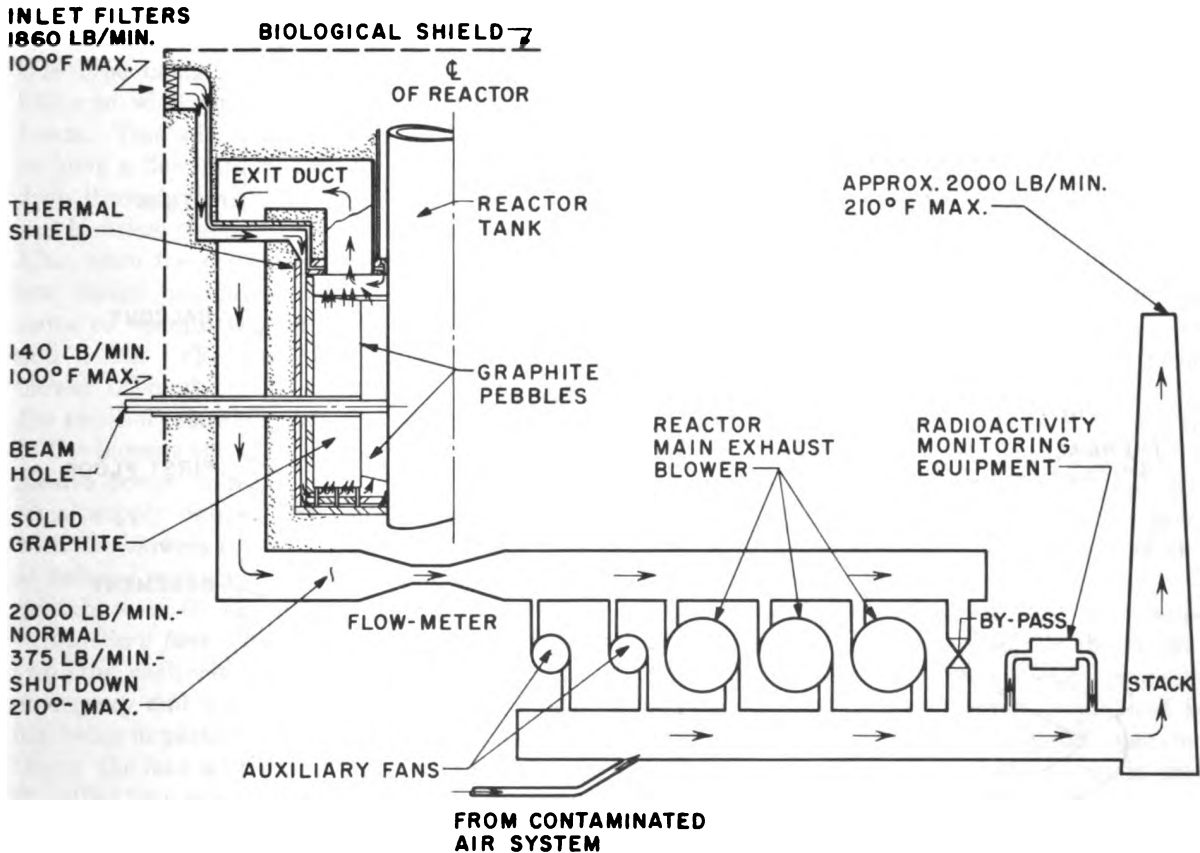


Fig. 3-26 Schematic flow diagram of reactor air system.

A separate channel with a damper is provided in each of the 8-in. ducts whereby a small amount of air is bypassed for cooling the top thermal shield. There is a total of twenty-two 8-in. ducts. Each duct has one 90° elbow and connects to the plenum chamber and the thermal shield manifold at 90° to the principal direction of flow. Thus there are effectively three 90° turns in the inlet air system. The walls of the ducts are lined with steel plate 1 in. thick to reduce gamma-ray leakage through the ducts.

The path followed by the air within the reactor

plenum chamber. The air from the chamber passes up through the graphite pebbles and cooling holes in the permanent graphite and is joined by air entering from the experimental facilities. The air enters a plenum chamber above the graphite where it is joined by the air bypassed to cool the top thermal shield. The air then enters the exit air ducts.

The exit air duct system within the reactor consists of two ducts 30 in. square connected to the thermal shield at diagonally opposite corners of the top plate and turned downward through the

biological shield to carry the air out below the basement floor. There are three 90° bends in each of these ducts, as shown in Fig. 3-27.

leads from this plenum chamber to a plenum chamber beneath the Blower and Fan House. The concrete duct is a minimum of 2 ft below

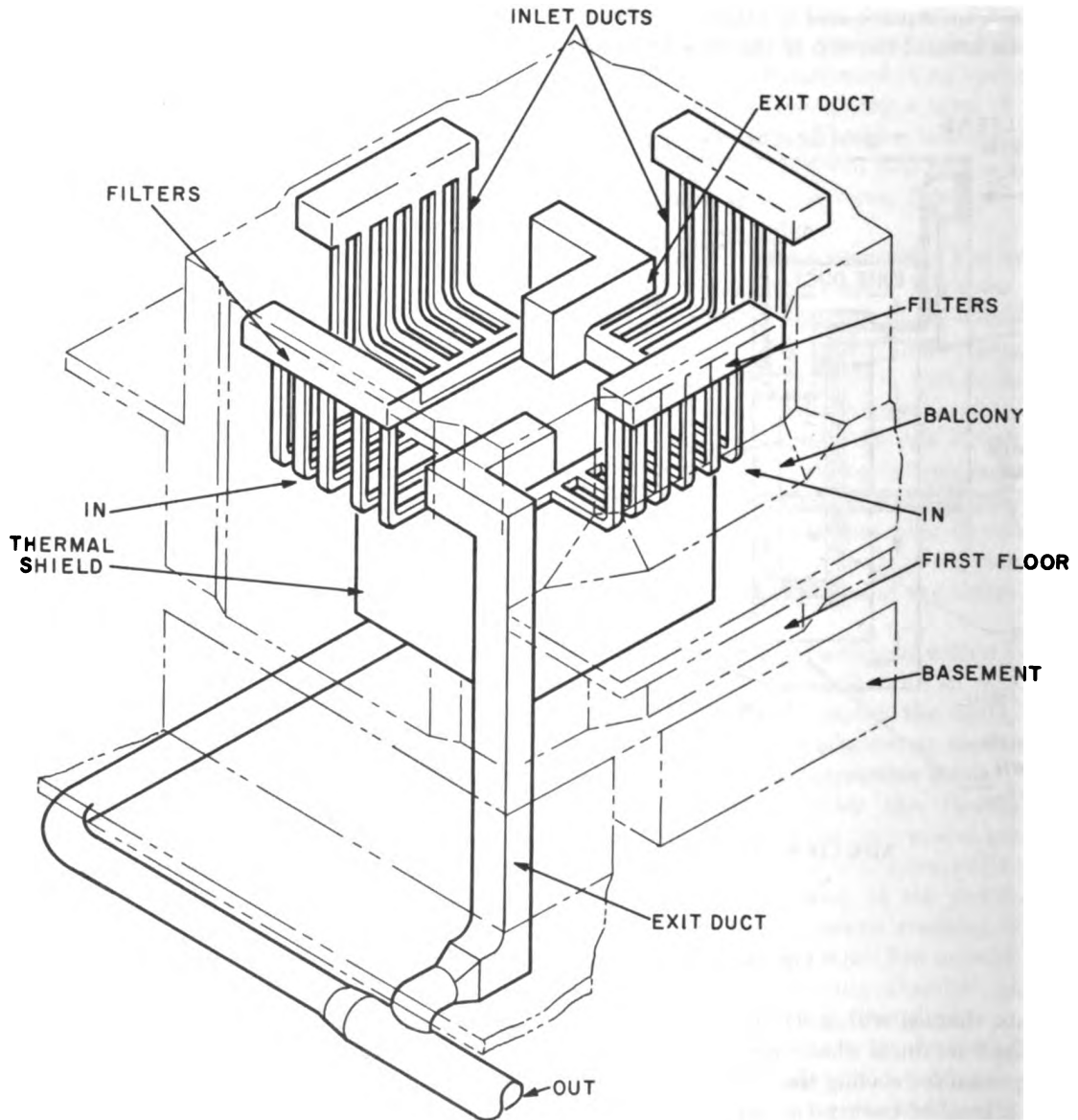


Fig. 3-27 Air system inside reactor structure.

Exit ducts. The 30-in.-square ducts leading down through the reactor connect to a single steel duct 48 in. in diameter beneath the basement floor. This duct connects to a plenum chamber outside the Reactor Building. An underground duct of precast concrete pipe 48 in. in diameter

grade so that the earth cover provides adequate shielding against radiation.

Blowers, fans, and auxiliaries. The blowers and fans of the reactor air system are housed in the Blower and Fan House and take their air from an inlet plenum chamber beneath the building

and exhaust it to another, also beneath the building.

Blowers. There are three half-capacity blowers in parallel, one of which serves as a spare, for cooling during normal operation of the reactor. Each blower is capable of delivering 1000 lb/min of 210°F air against a head of 55 in. of water and is driven by a 250-hp motor.

The blowers are of positive-displacement type, this type being selected because of its adaptability to wide ranges of flow rates and pressure heads. This is important since, while it is desired to have a flow rate of 2000 lb/min, the pressure drop through the system initially will be considerably below the design value of 55 in. of water. Also, since the cooling requirements may be below design calculations, it is desirable for the units to operate economically at reduced flows and heads. The positive-displacement type of blower offers the most economical operation at the reduced requirements.

The blowers are driven by electric motors which receive power from the normal commercial electrical supply only. Extra pulleys are provided with the blowers to provide for the slower speeds at reduced requirements. These pulleys will adequately cover the range from 700 to 1000 lb/min.

Auxiliary fans. Besides the blowers there are two fans, each with a capacity of 400 lb/min, for emergency and normal shutdown operation, each fan being in parallel with the three main blowers. One of the fans is driven by an electric motor and the other by a gasoline engine. The motor-driven fan receives its power from the normal commercial electrical supply and from the site emergency power supply and will be used during normal shutdown periods. The gasoline-driven fan will be used for emergency shutdowns when both electrical supplies have failed.

Valves. There are butterfly valves at both inlets and outlets of all blowers and fans. The valves on the inlet sides are manually operated, while the valves on the exhaust sides are air operated. The valves on the exhaust sides can seal airtight to permit removal of a blower or fan without outward leakage of radioactive air. The valves on the inlet sides do not have to be completely airtight since any leakage here will be directed into the reactor air system.

Since the blowers are of the positive-displacement type, it is necessary to have both the inlet and outlet valves open when the blowers are operating. To ensure this, limit switches are provided on both inlet and outlet valves. These switches require both valves to be open before the blowers can be started.

Also, because of the use of positive-displacement blowers, it was necessary to provide a bypass around all the blowers and fans so that the stack draft will be available to provide a flow of air through the reactor air system. The bypass is provided by a valve between the plenum chamber ahead of the fans and blowers and the exit duct leading to the stack. This valve will be operated manually.

Instrumentation and control. In the description of the instrumentation and control the flow path of the air will again be followed, starting with the inlets to the system.

There are four differential pressure indicators to determine the pressure drop across the glass-wool inlet filters. They are mounted locally on the pile faces. Each indicator determines the pressure drop across the filters on one side of the pile.

In the exit duct near the reactor is a pressure indicator to determine the pressure in the air system at that point. The indicator is remotely mounted in the Reactor Control Room and is equipped with an alarm to notify the operator when the pressure has risen above a certain predetermined value.

A venturi tube is mounted in the concrete pipeline leading to the Blower and Fan House to determine the flow rate through the system. The flow rate is indicated on a panel in the Blower and Fan House. Since the Blower and Fan House is an unattended station which has periodic inspections only, the information for operation of the reactor air system is transmitted to the Process-water Building and the Reactor Control Room. The Process-water Building serves as the center for operational information on the reactor air system. The flow rate is therefore recorded at the control panel in the Process-water Building, and there are low flow-rate alarms at both this panel and the panel in the Reactor Control Room. Besides the foregoing, the venturi information is

transmitted to the reactor control system to shut the reactor down automatically on failure of the air system.

In the exit air duct near the Blower and Fan House is a pressure indicator to measure the pressure in the system at this point. The information is indicated on a panel in the Blower and Fan House.

The valves on the inlet sides to the blowers and fans are manually operated. All of them are normally open, being closed only if necessary to repair or remove a blower or fan.

The main exhaust blowers are started manually within the Blower and Fan House. To make the operation of the reactor air system safer, both of the auxiliary fans start automatically on failure of the main blowers, and one may then be shut down manually. However, in the event of failure of the one left running, the other will automatically start again. The bearings on the main blowers and their motors are provided with thermocouples to indicate bearing temperatures and give an alarm at the panel in the Blower and Fan House. The high bearing temperature is also indicated on a master alarm at the Process-water Building panel.

A pressure indicator is located in the duct on the exhaust side of the blowers and fans. The pressure is indicated at the panel in the Blower and Fan House and is for operational information.

There are radiation-measuring instruments in the main duct leading to the stack. These instru-

In general, the contributions from the contaminated-air system are small or nil, and the measurements are essentially those of the reactor air system. The total activity of the gases to the stack is determined by an ion chamber, while that part due to active dust particles is determined by a particulate-activity meter. The information from both is transmitted to recorders in the radiation-instrument area of the Blower and Fan House. Alarms to indicate excessive activity going up the stack are provided at the Blower and Fan House, Process-water Building, and the Reactor Control Room panels.

Shielding. The shielding requirements of the reactor air system are based on an air flow of 30,000 cfm through the reactor at 60,000-kw operation. These assumptions gave shielding thicknesses which are adequate for all reactor operating conditions. To correspond to the 60,000-kw operating level, an average flux of 4×10^{12} neutrons/cm²/sec throughout the volume within the thermal shield was assumed. The volume of air subject to activation by this flux is approximately 850 ft³.

An examination of the elements normally occurring in air shows that it is necessary to consider only the active atoms N¹⁶, O¹⁹, and A⁴¹, which result from neutron capture by N¹⁵, O¹⁸, and A⁴⁰, respectively.

Table 3-5 summarizes the unit source intensities due to the active atoms, along with other pertinent information.

Table 3-5 Activity of Reactor Cooling Air

Original atom	Abundance in reactor air at 150°F, atoms $\times 10^{-23}$ /ft ³	Activation cross section, cm ²	Activation rate, atoms/sec/ft ³	Active atom	Unit source strength, Mev/sec/ft ³	Gamma energy, Mev
N ¹⁵	0.032	0.02×10^{-27}	2.56×10^5	N ¹⁶	3.12×10^5	6.5
O ¹⁸	0.0045	0.2×10^{-27}	3.5×10^5	O ¹⁹	7.8×10^5	1.6
A ⁴⁰	0.049	0.62×10^{-24}	7.2×10^9	A ⁴¹	4×10^6	1.37

ments are located toward the stack from the point where the line from the contaminated-air system enters this duct. Thus the instruments measure the total activities of the gases going up the stack.

Using the source intensities given in Table 3-5 the shielding requirements were determined with the use of standard shielding formulas (see Table 3-6).

Table 3-6 Shielding Requirements for the Reactor Air System

Item	Tolerance, r/8 hr	Shielding thickness	
		Concrete, in.	Steel, in.
Ducts in reactor building:			
2.5 ft square.....	0.005	11	
4 ft diameter.....	0.005	13.5	
Ducts outside:			
4 ft diameter.....	0.05	6.5	
Main blower cells:			
Outside walls (north and south) ..	0.005	4	1
Outside walls (east and west).....	0.005	8	
Intercell walls.....	0.05	*	
Floor above inlet plenum chamber	0.05	3	
Floor above outlet duct to stack ..	0.05	3	
Roof above cells.....	0.1		
Auxiliary fan cells:			
Floor above inlet plenum chamber	0.05	2	
Floor above exit duct.....	0.05	2	
All other walls.....		*	*

* Structural requirements only must be met.

Canal

The canal contains the equipment for the handling of assemblies discharged from the reactor and provides space for the storage of spent operating assemblies, irradiated materials, and internal parts of the reactor. Water over these materials protects the operating personnel from the hazards of radiation.

3-17 General Description. The general layout of the canal is shown by Fig. 3-28. The main section of the canal is 8 ft wide and extends eastward from the east face of the reactor. The canal section that lies partially beneath the reactor west wall is 6 ft wide. Connecting these two sections is a section, 7 ft wide, through the subpile room. The width of this section provides ample space for the canal unloading mechanism and storage of the neutron curtain.

The parapet around the canal is 10 in. thick at the bottom and projects outward at the top to a width of 13 in. This projection provides toe space and also gives the operator better stability when working over the parapet. The top of the parapet is 3 ft above floor level.

The bottom of the hydraulic shuttle canal is

6 ft below the basement-floor level and that of the main canal is 16 ft below the basement-floor level. The water level in the canal is maintained 2 ft above the basement-floor level, providing depths of water in the shuttle canal and main canal of 8 ft and 18 ft, respectively.

The depth of water in the canal is sufficient to shield adequately against the fission product gammas from the stored reactor-fuel assemblies. The main canal water depth of 18 ft limits the radiation levels to 0.01 r per 8 hr at the surface of the water in the reactor basement and to 0.1 r per 8 hr in the storage area.

Outside the Reactor Building the canal has a 6-ft wide working space on each side and at the end. To provide this working space there is a tunnel over the canal 13 ft 8 in. high and 21 ft wide. The tunnel extends 87 ft 6 in. beyond the Reactor Building east wall, 25 ft longer than had been planned originally, to provide additional storage space for spent assemblies.

The ceiling of the canal tunnel is constructed of reinforced concrete and is slightly below ground level. The section of roof below the road is reinforced to accommodate heavy trucks and the motor crane used to lift the spent-fuel assembly coffins. The unloading hatch is located in an offset widened portion of the road and is at a point where traffic has the least effect upon the unloading operation.

The portion of canal roof designed to withstand heavy loads is separated from the remainder of the roof by curbing which extends 50 ft along each side of the road. This prevents vehicles from traveling on the lightly constructed portion of the canal roof.

The canal walls are made of reinforced concrete. The canal floor is designed to withstand a load of 1 ton/ft²; the unloading sump can withstand a load of 2 tons/ft.²

The canal walls are covered with 8- by 16-in. white glazed structural tile 4 in. thick, and the bottom is lined with 4 in. of white concrete. This construction satisfies the requirements that there be good visibility in the canal and that cracks from which radioactive water could leak out of the canal be kept to a minimum. Other methods of construction, which were considered as satisfying either of these requirements, are listed below with reasons for their rejection:

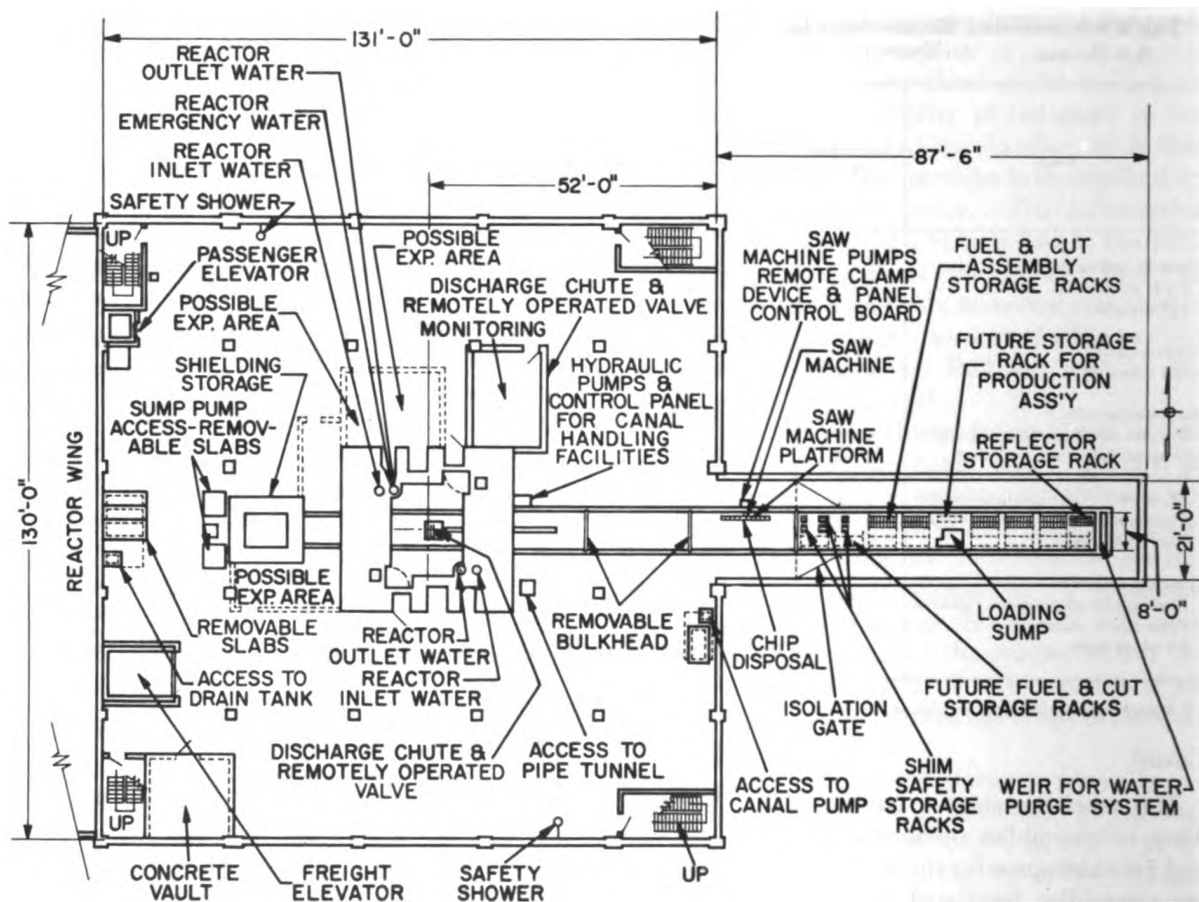


Fig. 3-28 Plan for Reactor Building basement.

1. Incorporation of a waterproof membrane in the canal walls to prevent outward leakage of water was discarded because gamma radiations from the spent-fuel assemblies in the storage area might cause deterioration and eventual failure of the membrane.

2. Painting of canal walls to enhance visibility was rejected because peeling and blistering would occur.

3. Covering the canal walls with a thin layer of white cement to aid visibility was rejected because of the likelihood of spalling.

4. The use of concrete wall surfaces was discarded because of their susceptibility to the accumulation of radioactive contamination.

The canal storage area is separated from the remainder of the canal by a bulkhead and gates. The underwater section consists of several removable bulkheads extending from the canal floor to

the water level. The section above the water is isolated by gates that swing from the walls and meet at the center of the canal. At the end of the canal the purge water discharges over a weir into the canal sump. Provisions are made for two additional removable bulkheads which permit division of the canal into three sections.

3-18 Canal Purge System. A flow of 100 gpm is maintained through the canal to avoid an excessive build-up of radioactivity and turbidity. A solenoid valve which automatically closes in case of power failure is installed in the well-water line at the inlet to the rabbit canal. This provision eliminates the possibility of flooding the canal sump when an electrical failure causes the canal sump pumps to stop.

Water from the canal overflows a weir, constructed the full width of the canal in the storage

section, and flows through a discharge pipe to the canal sump. The purge water from the canal sump is pumped to the acid-treating unit for use as cooling tower make-up water, to the leaching pond, or to the retention basin, depending on the activity level. A radiation alarm is provided in the sump, and manually operated valving for directing the flow is provided in the pump discharge header. A liquid-level controller governs the operation of the sump pumps.

3-19 Canal Flushing. Provisions are made for the installation of two removable watertight bulkheads, which permit isolation of any section, except the rabbit canal, for cleaning and maintenance. Each section except the rabbit canal is provided with a drain and a water-line connection. The drains are interconnected in such a way that any section can be isolated and flushed. There are three adjustable overflow standpipes in the subpile-room and rabbit-canal sections. A continuous overflow to the Reactor Building process sump is maintained through these in order to prevent scum formation.

3-20 Canal Lighting. Because the handling of assemblies in the canal is a manual operation carried on under 10 to 18 ft of water, good underwater lighting is necessary. This lighting is supplied primarily by lights suspended from portable fixtures attached to the side walls of the canal. In addition to general overhead lighting in the tunnel, special floodlights are provided in the loading-platform area, canal-storage area, sawing area, and loading sump.

3-21 Canal Crane. A manually operated crane with a 2-ton electric hoist is provided for the handling of heavy materials in the main canal. The highest hook position is 9 ft 9 in. from the floor. This provides a clearance of more than 6 ft between the hook and the top of the parapet. The hook cannot approach within 33 in. of one wall or 43 in. of the other wall of the canal tunnel. The maximum lift of the hoist is 25 ft 9 in.

Control System

3-22 General Description. The general scheme for controlling the reactor may be described as a combination of manual control by the operator

and automatic control from the instrument signals, so that, particularly during startup, the operator and the instruments serve as mutual checks on each other. Motion of the shim-safety rods to cause increased reactivity of the reactor requires the permission both of the operator and of various automatic devices and interlocks. However, an action tending to decrease the reactivity can be initiated by either the operator or several automatic devices independently. The action of the various scram circuits, i.e., those which shut the reactor down irreversibly in the event of an emergency, is entirely independent of the normal control and interlock circuits.

Stated in other words, the reactor is subject to effects of the control rods, auxiliary facilities, and certain disturbances. The reactor in turn affects certain instruments that produce information to be transmitted to the operator and to the control system. The operator and the control system also receive information from selsyns and other indicators of rod position or rod motion. On the basis of the information received, taken together with the operator's actions or instructions, the control system transmits to the motors or magnets (also called "clutches") appropriate control signals. The motors or magnets then cause corresponding rod motions, which in turn affect the reactor. Thus the main control loop is a closed path.

The two functions of the control elements are the safety function and the regulator function, the former being to guard against danger to the reactor and personnel and the latter to provide for startup and shutdown as well as for maintenance of a satisfactorily close regulation of the neutron flux over the required range of operating levels.

The safety and protective elements consist of those which warn the operator that action is necessary and those which automatically limit the speed of the control rod, reduce the power level, and shut the reactor down. The latter three actions are accomplished, respectively, by "period" control, a motorized rheostat, and release of magnetic clutches on the shim-safety rods. Period control is based on an electronic device that measures the time derivative of the logarithm of the neutron flux. It is used to prevent short reactor periods. Reduction of the power level to about

1 per cent of the normal operating level is valuable in emergencies in which the more drastic scram might not be justified and is brought about by signals not only from the instruments indicating neutron flux but also from the monitors on cooling-water activity, cooling-air activity, experimental facilities, instrument elements, etc. Scram signals come from instruments indicating the neutron-flux level and from a few critical monitors such as those for cooling-water flow. In addition to these devices a group of interlocks is used to serve as a step-by-step check on the operator to prevent serious consequences from mistakes or inadvertences.

The regulating requirements are met by a servomechanism which controls the motion of the regulating rod whose position in turn calls for the withdrawal or insertion of the shim rods. The regulating rod itself controls about 0.5 per cent of the reactivity (less than the effect of delayed neutrons) for purposes of fine regulation. The figure 0.5 per cent is chosen so that in the event of misoperation the servomechanism cannot change the reactor condition from critical to prompt critical. The shim rods and regulating rods are described in detail in Sec. 3-6.

Shim rods. A period meter has been developed which gives a scram signal if the period is less than 1 sec, and it causes the insertion of the shim rods by means of their motors when the period is less than 5 sec. This instrument is most useful below one-tenth of the full reactor power.

The design of the safety scram circuits is such that they give an alarm in the event that they are not in perfect operating condition. These alarms indicate not only failure but also incipient failure and where it occurs or is likely to occur (e.g., a short circuit in the cable, a weak tube, a lack of power).

Figure 3-9 shows the assembly of the reactor-tank system, including the top and bottom plugs, and the layout of the important reactor components.

Control Instrumentation

The neutron-flux signals for operating the control circuits come from 10 to 14 instruments located in the 10 instrument holes provided in the reactor structure. Three ionization chambers are reserved solely for the safety function to fur-

nish scram signals at 1.5 to 3 times the operating level and to furnish a slower signal to initiate reversible reduction of the operating level. Other signals provide either visual indication to the operator or signals to the servomechanism and other automatic devices. For visual indication from the operating level to 10^{-5} of the operating level, a direct-reading galvanometer is available. Another galvanometer shows the logarithm of the intensity factor from the operating level down to 10^{-11} of the operating level. In the lower part of the range the signal arises from a fission chamber and counting-rate meter.

There are three groups of instruments furnishing information to the operator and to the control system: (1) the reactor instruments measuring neutron flux or power level, (2) the rod-position and rod-motion indicators, and (3) the instruments for the auxiliary facilities.

3-23 Reactor Instruments. It is necessary to make a rather arbitrary distinction between reactor instruments and instruments on the auxiliary facilities, particularly on the cooling water. Reactor instruments include radiation instruments measuring the neutron level in the reactor and radioactivity in the water, temperature instruments measuring the distribution of exit-water temperature across the active lattice and temperatures in the thermal and biological shields, and pitot tubes measuring the distribution of water flow across the bottom (exit) of the fuel assemblies. The reactor radiation instruments are listed in Table 3-7.

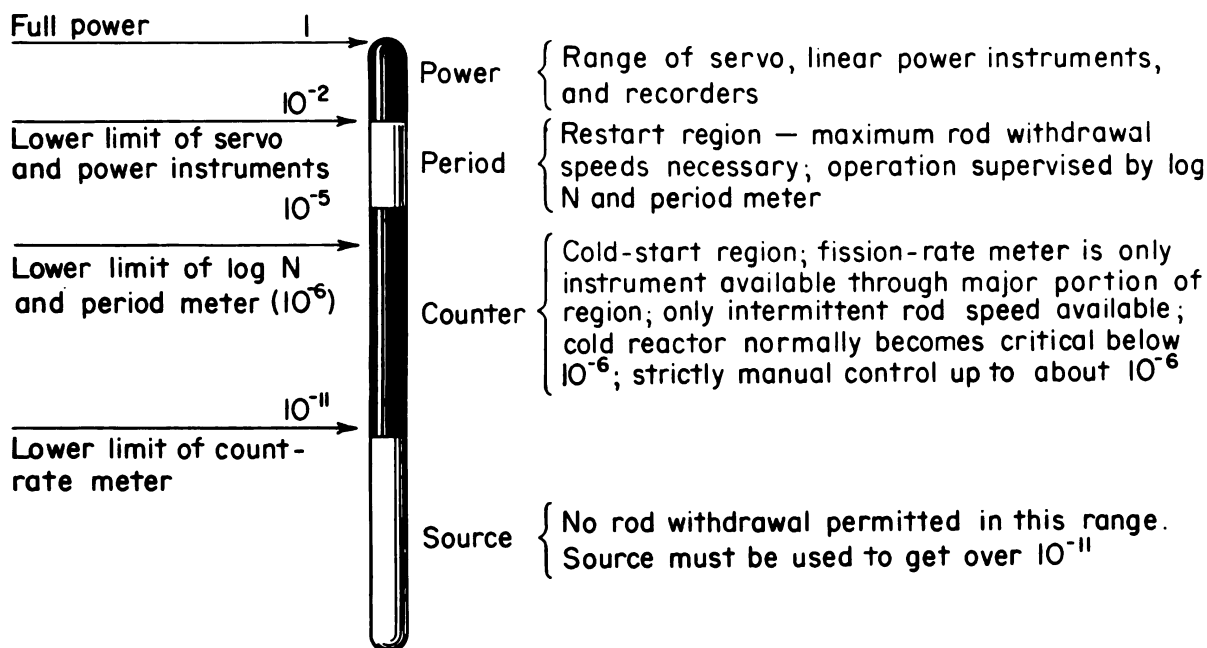
Table 3-7 Reactor Radiation Instruments

<i>Instrument</i>	<i>Use</i>
Parallel-circular-plate ionization chamber.....	Safety and servo signals
Compensated ionization chamber.....	$\log N^*$ and period signal; galvanometer signal
Fission chamber, 12-ft travel.	Counting-rate signal
Air-wall ionization chamber..	Water-flow N^{16} monitoring
Water-monitor ionization chamber.....	Water-sample monitoring
Boron thermopiles.....	Neutron flux in graphite

* N here means essentially neutron flux (nv).

The importance of having proper instruments for control purposes is made evident by considera-

Table 3-8 Relation of Power Level to Instruments



tion of Table 3-8, which shows the relation of the power (or neutron) level considered in four definite ranges to the choice of instrument and control action. These ranges serve as convenient divisions of the entire range of reactor operation. At the lowest level is the source range, which lies below the range of the counting-rate meter and in which control-rod withdrawal should be prohibited. Until the level is raised above 10^{-11} of full power by a suitable source, the effective k of the reactor should be kept less than 1.0, i.e., rods should not be withdrawn. Next comes the counter range, in which the combination of fission chamber and counting-rate meter is the source of a reliable signal. For the MTR, only intermittent rod withdrawal is permitted in this range. The lower limit of the period meter makes the transition to the period range. By the time this range is reached, the cold, clean reactor will have become critical and the level will be rising on a stable period. Finally comes the power range, over which an appreciable fraction of full power may be measured.

3-24 Parallel-circular-plate Chambers. The parallel-circular-plate (PCP) ionization chambers

are designed, constructed, and located in the reactor so as to give high-speed response to changes in neutron flux coming from the core of the reactor. The active section consists of a set of graphite disks, each of which is coated on both sides with B^{10} . Alpha particles are emitted by the $B^{10}(n, \alpha) Li^7$ reaction when neutrons bombard these disks. These alpha particles cause the ionization which the chamber measures. The thickness of the boron coating (0.3 mg/cm^2) on the graphite disks is chosen so as to give the maximum ionization current (about $50 \mu\text{a}$ at their operating flux of 10^{10} neutrons/cm²/sec).

The chambers cannot be located as near the reactor tank as would be desirable because of the high neutron flux. They are located in holes, looking directly toward the tank, with collimating boral sleeves ahead of the chambers to enhance their inherent directional sensitivity to flux directly from the tank. With this arrangement the major delay is the time, estimated at 5 to 10 msec, for neutrons to get from the core through the tank and into the graphite.

3-25 Compensated Ionization Chamber. The compensated ionization chamber is designed

to give reliable measurement of neutron flux over a large range, particularly in the presence of relatively intense gamma radiation. The construction features two separate chamber volumes. The inner volume is contained between a movable cup electrode and a fixed inner electrode shell; the outer volume is between this inner shell and an outer electrode shell. The two volumes being approximately equal, the effects of gamma radiation on the two should also be equal. In addition, the outer volume is made sensitive to neutrons by B^{10} coatings applied to the electrode surfaces. Then, if the cup electrode is at negative potential and the outer-shell electrode is at positive potential with respect to the inner shell, the net current carried by the inner shell will be a measure of neutrons alone. Close balancing of the two volumes for zero gamma signal is carried out by moving the cup electrode, thus varying the inner volume.

3-26 Fission Chamber. The fission chamber is designed to give neutron measurements at low flux levels. The construction is quite simple. A brass shell carries a nickel liner with a coating of U^{235} on its inside surface. A central anode rod of 2S aluminum, 0.25 in. in diameter, is supported on a coaxial connector. Neutrons absorbed in the U^{235} produce fission fragments which ionize the gas in the chamber. The voltage pulse thus created by each fission is amplified and counted. These fission pulses are large and, by proper biasing of the amplifier, can be counted separately from the lesser currents due to alpha or gamma radiation; i.e., neutron flux can be accurately measured even in the presence of intense gamma radiation. By using argon gas, a short, sharp pulse may be obtained. However, the chamber performance is then very sensitive to gas impurities and to supply voltage; to minimize these difficulties, the argon is deliberately mixed with about 5 per cent carbon dioxide. Under these conditions, pulses are distinguishable up to a limiting rate of nearly a million per second. The counting-rate meter used with the fission chamber, however, has a useful range from 1 to 10^4 counts/sec, and this is therefore the working counting range of the complete instrument.

To increase the neutron-flux range of the instrument, provision is made for moving the cham-

ber over a maximum distance of 12 ft, from outside the top thermal shield to a point well inside it. This travel corresponds to an estimated change in sensitivity of 10^7 . At the innermost position the minimum flux to be expected (for example, from a source at the startup of a clean reactor) would be about 10^3 neutrons/cm²/sec, corresponding to 10 counts/sec. The maximum flux (reactor at full power) at the inner position would be 10^{13} neutrons/cm²/sec, and at the outer position, about 10^6 neutrons/cm²/sec, corresponding to 10^4 counts/sec, the upper limit of the counting-rate meter. In view of this range the fission chamber is used at three established positions in its instrument hole, motion from one to the other giving a neutron-flux range multiplication of about 3×10^3 , which is comfortably within the counting-rate range of 10^4 counts/sec. The means of moving the chamber is semiautomatic and is connected with the control circuit.

3-27 Air-wall Ionization Chamber. The air-wall ionization chamber is used to measure the energetic 7-sec gamma activity of N^{16} formed in the $O^{16}(n,p)N^{16}$ process by neutron bombardment of the cooling water. The chambers are located in thimbles in the outlet flow. This signal should constitute a good over-all measure of the average reactor core flux. The chamber is 5 in. in diameter by 19 in. long, with cylindrical electrodes. The name "air wall" signifies that it is made of materials having nearly the same atomic number as air and hence giving similar scattering.

3-28 Water-monitor Chambers. The water-monitor chambers are used in the monitoring room to measure beta and gamma activity in the water drawn through the sampling tubes from the exit face of the active lattice, one tube under each possible element position. Thus the distribution of radioactivity in the water flowing through the lattice can be observed. Water from each of the tubes is examined every 2 min by an automatic cycling device. The water is held up long enough for the N^{16} activity to become small; therefore the observed activity is either that induced by the flux in the demineralized cooling water or it is evidence of a fission break (rupture of a fuel element) in the core.

3-29 Boron Thermopiles. The boron thermopiles are used for an additional check on the neutron-flux level. They are all located in the graphite at the same horizontal level. These instruments consist of a large number of thermocouple junctions in series, alternate junctions being coated with boron. The coated junctions become warm owing to absorption of neutrons, thus yielding a net voltage which is a measure of neutron flux. The instrument case is about 0.5 in. in diameter and 6 in. long.

3-30 Other Instruments. In addition to the foregoing reactor instruments, all of which measure radiation, there are various temperature and water-flow measurements to be mentioned. Each of the sampling tubes under the fuel element positions is actually part of a triple-function assembly: sampling tube, pitot tube for flow measurement, and thermocouple for water-temperature measurement. Knowledge of the distribution of water flow, velocity, and temperature is of obvious value. There are also numerous thermometric devices used for observing temperatures in other parts of the reactor. Most important of these, from the control standpoint, is the pair of thermocouples in the inlet and outlet water lines, measuring the temperature rise which is then multiplied by total water flow to give power absorbed from the reactor. A venturi flowmeter feeds the signal to a flow recorder, and the thermocouples feed a differential temperature recorder. Impedances in these recorders are connected in a bridge circuit in such a way that the bridge output is proportional to the product of the two signals. The detector for this bridge is thus the power-level recorder.

In the thermal and biological shields there are about 50 positions for thermocouples. The signals from these instruments, however, are connected with the control of the reactor only through the one annunciator channel reserved for them.

Control Console

3-31 The control console, located in the reactor control room, is made up of five unit panels laid out on a circular arc of 51 in. radius and extending some 240°. At the center of this arc, the operator is seated in a movable chair, giving him easy

access to the information and control elements spread out on the panels. At his left hand is a telephone by which he can communicate with other control and instrument centers, such as the one in the Process-water Building. By looking over the top of the console, which is 40 in. from the floor, he can see instrument panels standing in an arc directly in front of him.

On the outside of the console, each of its five units has vertical-hinged doors to give free access to the interior (see Fig. 3-29). Each panel is either hinged or otherwise removable so as to give access from the top.

Figures 3-29 and 3-30 also show the various indicating and control devices mounted on the panel. These items will be briefly mentioned here.

The panels will be discussed as faced by the operator, from his left to his right. On his extreme left is panel *N*, carrying eight identical units, one for each of the eight possible shim-rod positions. Each unit contains two principal elements, a selsyn receiver dial and a switch. The selsyn dial is calibrated in inches from 0 to 30 with clockwise rotation indicating withdrawal. Associated with the dial are four amber lights which come on, respectively, when the magnetic clutch is not holding its rod, when the upper switch is actuated. That is, in normal operation, all four lights would be off. The switch has six positions, giving different controls over rod motion. On either side of the switch are amber lights, the left indicating that a rod is selected for withdrawal, the right that it is preferred for insertion. Also on the left is a red light indicating that withdrawal is in progress and on the right a similar green light for insertion. Finally, a name plate identifies the shim rod by number.

Next is panel *P*, carrying the selsyn receivers for the regulating rods and also a regulating-rod selector switch. These selsyn dials are also calibrated in inches with clockwise pointer rotation indicating withdrawal. The 22 in. of possible rod motion is spread out for extra sensitivity over one and a quarter pointer revolutions so that the first few inches and the last few overlap. Amber lights mounted under the scale come on to indicate actuation of the regulating-rod limit switches.

Panel *Q*, at the center of the console, is devoted to two of the most important instruments,

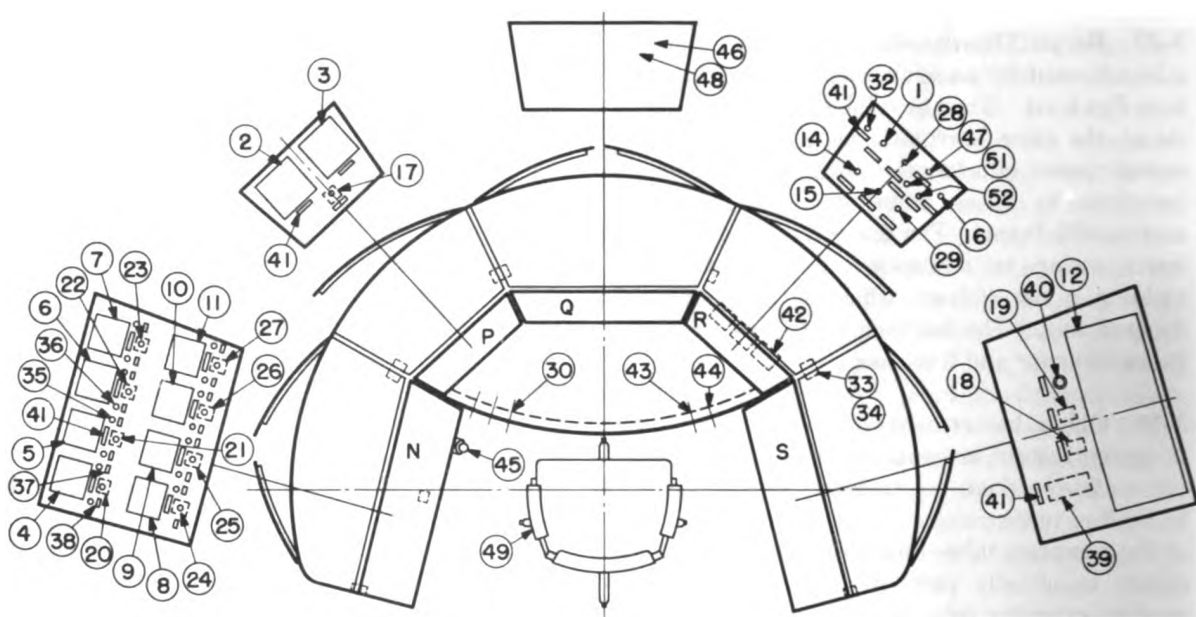


Fig. 3-29 Control-console layout.

Control Console Instrument Schedule

Item	Instrument	Service	Item	Instrument	Service
1	Reactor period	Indicator	29	S2B switch	Shim rods inter-high speed
2	Selsyn reactor	Reg. rod No. 1 pos.	30	SPST switch	Console ceiling lights
3	"	Reg. rod No. 2 pos.	32	4PDT switch	Period channel selector
4	"	No. 1 shim rod pos.	33	Wire terminal blocks	Conductors
5	"	No. 2 shim rod pos.	34	Wire fanning strips	Conductors
6	"	No. 3 shim rod pos.	35	Lights 12V "amber"	No. 1 to No. 8 shim rods preferred
7	"	No. 4 shim rod pos.	36	Lights 12V "amber"	No. 1 to No. 6 shim rods selected
8	"	No. 5 shim rod pos.	37	Lights 12V "green"	No. 1 to No. 6 shim rods insert
9	"	No. 6 shim rod pos.	38	Lights 12V "red"	No. 1 to No. 6 shim rods withdraw
10	"	No. 7 shim rod pos.	39	S6 push button	OFF-START-RUN operation
11	"	No. 8 shim rod pos.	40	Vernier	Set point
12	Procedure panel	Indication	41	Name plates	Identification
14	S1 switch	Shim-rod reverse by-pass	42	Transformers	Indicating lights
15	S2A switch	Shim rods insert and withdraw	43	SPST switch	Annunciator reset button
16	S3 switch	Shim rods selector	44	SPST switch	Annunciator acknowledge
17	S4 switch	Reg. rod selector	45	Phone	Communication
18	S5 switch	Shim rods raise clutch	46	Galvan. panel	Part of galvanometer
19	S7 switch	Reactor level set point	47	SZ push button	Shim rods scram reset
20	S21 switch	No. 1 shim-rod control	48	Galvanometer	Indicates level and counting rate
21	S22 switch	No. 2 shim-rod control	49	Chair	Console contr. operator
22	S23 switch	No. 3 shim-rod control	51	Light 12V "red"	SLOW-SCRAM indicator
23	S24 switch	No. 4 shim-rod control	52	Light 12V "red"	SLOW-SCRAM indicator
24	S25 switch	No. 5 shim-rod control			
25	S26 switch	No. 6 shim-rod control			
26	S27 switch	No. 7 shim-rod control			
27	S28 switch	No. 8 shim-rod control			
28	SX switch	Shim rods manual scram			

the flux-level galvanometer and the counting-rate galvanometer. These instrument signals are given in a pair of glass scales at the top of the panel, the upper one for flux level and the lower serving either for counting rate or for level deviation.

marks, an upper set constituting four decades of a logarithmic scale for counting rate and a lower set of linear marks from -250 to $+250$ for the level-deviation readings. The galvanometer serving this scale can be switched by a "select-circuit"



Fig. 3-30 Control-console operation.

The flux-level scale shows the signal coming from the compensated ion chamber. To give full range to this signal, two galvanometers, one of medium and one of high sensitivity, are used, each with three different decade values of Ayrton shunt. The total spread in these six ranges is thus 10^5 . This choice of range is made on a flux-level range switch mounted in the lower right-hand corner of the panel.

The lower glass scale carries two sets of scale

switch next to the flux-level switch to "counting" (in which case its signal comes from the fission chamber) or to one of three (low, medium, high) sensitivities for the flux-level signal. This is the "deviation" measurement, which is useful for fine manual control when the level is high and the flux-level galvanometer itself has therefore been cut back to low sensitivity.

Panel Q also carries a number of knobs for adjustment of zero, calibration, and adjustment of

balance voltage for the galvanometers. There are, moreover, two additional items which deserve particular mention. One is the set of three (out, stop, and in) push-button controls in the lower left-hand corner for movement of the fission chamber. The other is the pair of alarm lights and reset buttons at the upper right-hand corner which work from phototube alarm circuits when any of the galvanometers go off scale.

To the right of panel *Q* is panel *R* with two (horizontal) rows of elements. From left to right on the upper row are a double-throw switch for selection between the two available compensated chamber period channels, the period indicating meter, the manual scram switch, the shim-rod withdrawal selector switch, and the scram reset push button. The selector switch is an eight-position switch for selection of any one of the shim rods to be withdrawn. On the lower row are three more switches relating to shim-rod control. On the left is the reverse switch labeled *WITHDRAW*, *NEUTRAL*, and *INSERT*. On the right is the intermittent- and high-speed switch labeled *INTERMITTENT*, *NEUTRAL*, and *HIGH*.

On the extreme right of the operator is panel *S*. This is devoted largely to the procedure panel, which extends across the top. The procedure panel is an assembly of lights (green with one exception) with labels by means of which the over-

all status of the control system is reflected. Underneath the procedure panel there are, from left to right, the vernier potentiometer for the reference voltage on the servoamplifier input; the reactor level set-point switch, which controls the motor-operated rheostat; the switch controlling raising of a shim-rod holder when there is no rod on it (e.g., during maintenance in dry dock) and, in its other position, for starting an amplidyne to cock a regulating rod; and three push buttons (off, start, run) for over-all control, together with a cylinder lock requiring a key for starting.

Control-room Instrument Panel

3-32 The control room also includes an instrument panel to accommodate the indicators, recorders, and annunciators with a number of spare and partially utilized panel sections for future requirements. This panel board, facing in the direction of the reactor, is composed of 11 sections 4 ft wide and 9 ft high, and extends the full length of the room with a rear access door at each end. Six panel sections, three at each end of the board, lie in a straight line. These two end panel groups are connected by the central panel sections which are arranged on a radius of approximately $7\frac{1}{2}$ ft to allow for the control console.

PART 2 DESIGN CALCULATIONS

This part summarizes the physics calculations and experiments that supported the design of the MTR.

Sections 3-33 to 3-36 describe the Oak Ridge critical experiments that led up to the MTR, including the mock-up. The associated calculations are mentioned. Sections 3-37 and 3-38 concern the reactor-theory computations for the MTR itself; Sec. 3-39, the heat production, transfer, and removal; and Sec. 3-40, the shielding calculations for the reactor, pipes, etc.

OAK RIDGE CRITICAL EXPERIMENTS AND CALCULATIONS

3-33 First MTR-like Critical Experiment.

The first critical experiment with an assembly that closely simulated the final MTR design was done in 1947. As in the MTR, the core was composed of enriched uranium, Al, and H_2O . The reflector was a thick layer of Be but contained no water to simulate the coolant as did the later criticals. The ratio of Al to H_2O was 0.76 by volume (the ratio Al/ H_2O in the first MTR core, including one-half the water between core and reflector, was 0.732).

The core of the critical experiment was built up with Al tubes, 1 by 1 by 35 in. long, filled with fuel solution having a concentration of U^{235} of 35.6 g/liter of reactor. The height of the active portion was 70 cm. The reactor went critical four full tubes short of a 9×9 square array; so a number of uniformly distributed gold wires 0.030 in. in diameter were inserted to make the reactor critical with 1.54 kg of U^{235} in the full square array (25.8 by 25.8 cm). However, by correcting for the empty control-rod ports and gold poison it was estimated that the minimum critical mass for such a square array would be 1.42 kg.

Calculations were made on this experiment using a two-group two-region model, and the critical mass was calculated to be 1.54 kg. This was only 8.5 per cent too high, which is good agreement. The calculated value of 1.54 kg was later reduced to 1.47 when it was realized that the assembly was not really bare on the top and bottom be-

cause of the presence of rubber stoppers in the fuel tubes as well as other structural material.

The calculation was made using the slab-geometry approximation and "reflector-savings" extension of the core in one direction since the parallelepiped geometry is only soluble with reflectors on one pair of faces. The constants used are given in Table 3-9.

Table 3-9 Constants of First MTR-like Critical Experiment

Region	r , cm ²	D_f , cm	L^2 , cm ²	D_s , cm
Core.....	65.6	1.26	3.65	0.267
Reflector.....	94.2	0.630	625	0.772

It was concluded that "with sufficient knowledge of the experimental conditions, two-group theory gives reliable results for the critical masses of enriched water-moderated piles." The qualification in this conclusion was pertinent since it now appears that the principal reason for disagreement between theory and experiment with the MTR itself was the presence of Li poison in the fuel-element welding flux—an experimental condition about which there was no knowledge.

3-34 Two Further "Clean" MTR-like Assemblies. Two further Be-reflected critical experiments were also done in 1948. The cores were like the MTR but had an Al/ H_2O ratio of 0.65. Again there was no simulation of the reflector cooling water.

Of interest also in this group of experiments was an unreflected assembly of Al/ H_2O ratio, 0.65. A layer of Cd was wrapped around the bare pile to reduce backscattering from the floor, etc. Criticality was not quite obtained, but extrapolation of the curve of reciprocal counting rate yielded an estimate of 5.3 ± 0.1 kg. The dimensions of the pile with the largest k_{eff} that was assembled (U^{235} weight of 5.05 kg) was 66 by 48 by 43 cm.

The first of the pair of Be-reflected cores was square in horizontal cross section and bare on the

top and bottom. The Be was more than 30 cm in thickness around the vertical sides. The Al/H₂O ratio was 0.65. The height of the core was 66 cm, and a space within to the stacked Be of dimensions 22.5 × 22.5 cm was left for the core. This allowed room for an 8 × 8 array of fuel tubes. The assembly went critical just three fuel tubes short of the 8 × 8 array, and about 60 cm² of uniformly distributed gold wires was inserted in order to attain the full loading. The full 66- by 22.5- by 22.5-cm reactor with 60 cm² poison then contained 1.21 kg of U²³⁵ at a fuel concentration of 38 g U per liter. When corrections were made for the control-rod ports and the 60 cm² poison, it was estimated that the minimum critical mass of such a square cross-section assembly with thick Be reflector would be 1.07 kg.

The second Be-reflected core was slablike in cross section and again bare at the top and bottom with more than 30 cm reflector around the sides. The height was 66 cm, and the reactor went critical with a horizontal cross section of 11 by 51 cm with 30 cm² of uniform extra poison. The mass of U²³⁵ was 1.35 kg, but it was estimated that the minimum critical mass for such a slablike geometry would be 1.28 kg when corrections were made for the control-rod ports and extra poison.

A calculated result for the latter assembly was obtained by using the experimental uranium mass as well as the experimental size and determining the necessary k_{∞} (the infinite pile multiplication constant) for criticality. The calculated k_{∞} was 1.598 as compared with the experimental k_{∞} of 1.576 (the physical k_{∞} of the core mixture). This means that if the calculation had been made in terms of critical mass, the calculated critical mass would have been too high by approximately 6 per cent. The constants used in the calculation are not available other than the extrapolation distance of 8.1 cm at the top and bottom. This is an effective extrapolation distance and is larger than it would be for a truly bare surface due to the corks at the ends of the fuel tubes.

3-35 A Series of Criticals Approaching the MTR Design in Detail. The critical experiment program was continued in 1948 with several assemblies that were less "clean" than those just described but each progressively closer to the

actual MTR design. The starting point was the assembly discussed above, having a height of 66 cm, a horizontal cross section of 51 by 11 cm, a 30-cm Be reflector on the vertical faces and no reflector at the top or bottom, an Al/H₂O ratio of 0.65, and a final concentration of 37.1 g of U²³⁵ per liter of core. This was called assembly 2.

The length was then increased to 71 cm for assembly 3a. This required the uniform addition of 226 cm² of poison to hold it down to just critical, where this poison was in the form of $\frac{1}{8}$ - by $\frac{15}{16}$ -in. strips of type 347 stainless steel inserted in the place of Al.

Thirty centimeters of graphite was now stacked on one side outside the Be to make assembly 3b which required 303 cm² of poison to keep it to just critical.

The width was now increased from 11 to 14 cm, keeping everything else the same. This was called assembly 4, and it required 640 cm² to prevent it from exceeding criticality.

The width was then increased again to 17 cm, making the dimension 71 by 17 by 66 cm. This was assembly 5, requiring 1140 cm² to prevent it from exceeding criticality.

The width was now increased a third time to 22.5 cm, and Lucite was added uniformly to the Be reflector to simulate 2 per cent cooling water (the MTR as it was finally constructed has 3.2 per cent water in the L pieces, 5.6 per cent water in the 2-in. width of permanent reflector adjacent to the active lattice, and 1.7 per cent water in the 5-in. width of permanent reflector adjacent to the 2-in. region). This assembly 6 required 1580 cm² of poison to prevent it from exceeding criticality.

The final critical assembly was as realistic a mock-up of the MTR design of that date as could be made, in so far as material composition and dimensions were concerned. Seven 6-in. holes were put in the Be reflector, and the graphite stacking was continued until 30 cm of graphite was on all sides of the Be. This assembly 7 then had 3.95 kg of U²³⁵ and 1580 cm² of uniform poison, making $\Delta k/k_f$ excess of 18.8 per cent with k_f (final k_{∞}) of 1.3603 (the initial $k_0 = 1.6167$).

The principal differences between this assembly 7 and the MTR as it was actually constructed were as follows: (1) The Al/H₂O ratio in the core was 0.65 in the critical experiment as compared to 0.732 in the MTR. (2) The Be reflector of the

critical experiment had seven 6-in. holes, four of them beginning at the core and three of them beginning 15 cm out from the core, whereas the MTR when in the condition at which criticality determinations are normally made has dummy Be plug tips in all the beam holes; on the other hand the MTR has (a) three 2-in. DB channels beginning at the core on the north and three 2-in. DB channels beginning 15 cm out from the core on

assumes that similar quantities of chlorine were left in the elements, then the $\Delta k/k_{eff}$ could have been more than 3 per cent. A brazed fuel element was later compared against an unbrazed one in the swimming-pool reactor, indicating again that the brazing poison probably reduced the reactivity of the MTR by about 3 per cent.

Table 3-10 summarizes the results of the seven criticals just discussed.

Table 3-10 Critical Mass and Poison Content (Excess k) of U-A1-H₂O Reactors with Be Reflector

A1/H₂O (volume) = 0.65; k_0 (clean k_{∞}) = 1.6167; Be ~ 30 cm thick—reactor bare top and bottom

Assembly	Dimensions	Mass, kg	Σ poison, cm ²	Σ pile, cm ²	k_f	Excess reactivity, %	
						$\Delta k/k_f$	$\Delta k/k_0$
1	22 × 22 × 66	1.21	66	2591	1.5626	3.5	3.4
2	51 × 11 × 66	1.35	30	2882	1.5849	2.0	2.0
3a	71 × 11 × 66	1.94	266	4118	1.5184	6.5	6.1
3b	71 × 11 × 66	1.94	303	4118	1.5057	7.4	6.9
(30 cm graphite on one side)							
4	71 × 14 × 66	2.44	640	5179	1.4388	12.4	11.0
(30 cm graphite on one side)							
5	71 × 17 × 66	2.94	1140	6240	1.3640	18.3	15.5
(30 cm graphite on one side)							
6	71 × 22.5 × 66	3.95	1848	8384	1.3247	22.0	18.0
(30 cm graphite on one side; 2% water in Be)							
7	71 × 22.5 × 66	3.95	1580	8384	1.3603	18.8	15.8
(30 cm graphite on all sides of Be. Seven 6-in. holes in reflector; 2% water in Be)							

the south, (b) a 5- by 12-in. channel (HG-9) beginning 15 cm out from the core on the east face, and (c) two 1-in. pneumatic rabbit holes (HR-1 and HR-2) running north and south 1 in. from the east face embedded in considerable aluminum. (3) The critical experiment had less water in the Be adjacent to the core than does the MTR. (4) The first MTR fuel elements contained a significant amount of Li poison from the brazing flux which was not present in the critical experiment. A chemical analysis of the brazing flux on a typical MTR fuel element showed that it contained 1.26 g of Li. This amount of poison in all elements would have a $\Delta k/k_{eff}$ of -2.7 per cent, and if one

The calculations on assembly 2 have been previously discussed. Calculations were also made on assembly 3 by using the experimental size as well as the experimental uranium mass and determining the necessary k_{∞} for criticality. The calculated k_{∞} was 1.537, whereas the experimental k_{∞} was 1.528 (this differs slightly from the experimental k_{∞} given in ORNL-167). The difference (1.537 vs. 1.528) indicates that if the calculation had been made in terms of critical mass the calculated critical mass would have been too high by approximately 4 per cent. The constants used are not given other than an effective extrapolation distance at top and bottom of 8.1 cm.

3-36 The MTR Mock-up. In the spring of 1947 the Kellex Corporation started designing a full-scale mock-up of the MTR, the original purpose of which was to conduct mechanical and hydraulic tests without neutrons. The mock-up in the form in which it was constructed for these tests had an aluminum reflector as well as other significant differences from the MTR design. When the mechanical and hydraulic tests had been completed, it was decided to modify the machine such that it would resemble the MTR design as closely as possible and to operate the mock-up as a low-power reactor.

The modifications made in the mock-up at this time were begun on Nov. 5, 1949, and completed in Jan. 20, 1950.

The essential differences between the mock-up as a critical assembly and the MTR design are as follows: (1) the tank on the mock-up is $\frac{3}{4}$ -in. 3S aluminum whereas the MTR tank is 1-in. 2S aluminum; (2) the HB holes in the mock-up are 4 in. below the center of the fuel; (3) the Be reflector of the mock-up was only on the north and west faces of the lattice. Because of the size of the pieces it was not possible to obtain a close fit at curved surfaces. One row of L pieces surrounded the fuel pieces in lattice positions. The rest of the space in the tank was filled with water.

Criticality was attained in the mock-up on Feb. 4, 1950. The critical mass was 2020 g contained in 13 fuel elements. The loading was in the shape of a slab arranged three assemblies wide by five assemblies long, including two safety rods.

The geometry of the reflector configuration is so asymmetrical that it would be virtually impossible to make a critical-mass calculation that would be of any real value for correlation purposes.

REACTOR-THEORY COMPUTATIONS FOR THE MTR DESIGN

3-37 Flux Calculations. One of the most important reasons for using a small-volume light-water-moderated core instead of a large-volume heavy-water-moderated core was that the fast flux and the gamma-ray flux are much higher. Table 3-11 shows the comparison that was calculated.

Table 3-11 Fluxes in Light-water and Heavy-water Reactors

	Light water	Heavy water
Powers, kw.....	30×10^3	30×10^3
Power/volume, kw/liter..	357	37
Maximum virgin nv	$\sim 1.2 \times 10^{14}$	$\sim 0.15 \times 10^{14}$
Maximum epithermal nv	$\sim 6 \times 10^{14}$	$\sim 5 \times 10^{14}$
Maximum slow nv	$\sim 2 \times 10^{14}$	$\sim 2 \times 10^{14}$

A second desirable characteristic of the small-core light-water-moderated reactor is that the thermal-neutron-flux distribution is essentially flat in the core and for a short distance into the beryllium reflector is equal to or greater than the average core flux.

The magnitude of the rise in the reflector depends on the size and shape of the reactor. In Figs. 3-31 to 3-33 the radial flux distributions as

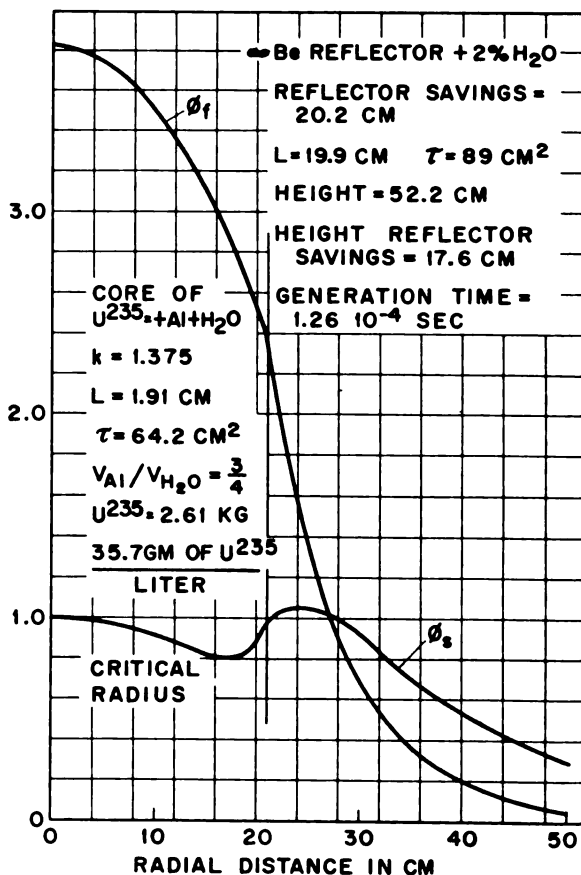


Fig. 3-31 Neutron-flux distributions.

calculated for three cylindrical reactors with different k values are shown. In the largest reactor (lowest $k = 1.373$) the thermal-flux rise in the reflector is least pronounced. The larger the reactor, the lower the rate of thermalization, which is proportional to ϕ_f (flux above thermal), becomes in the reflector. This results in a decreased ϕ ,

in the beryllium 5 cm from the reactor surface; at the beryllium-graphite interface, it would be about 6×10^{13} ; and at the thermal shield, about 3×10^{11} . As it turned out, the flux rise in the reflector is considerably greater than was predicted by two-group theory; fluxes as high as

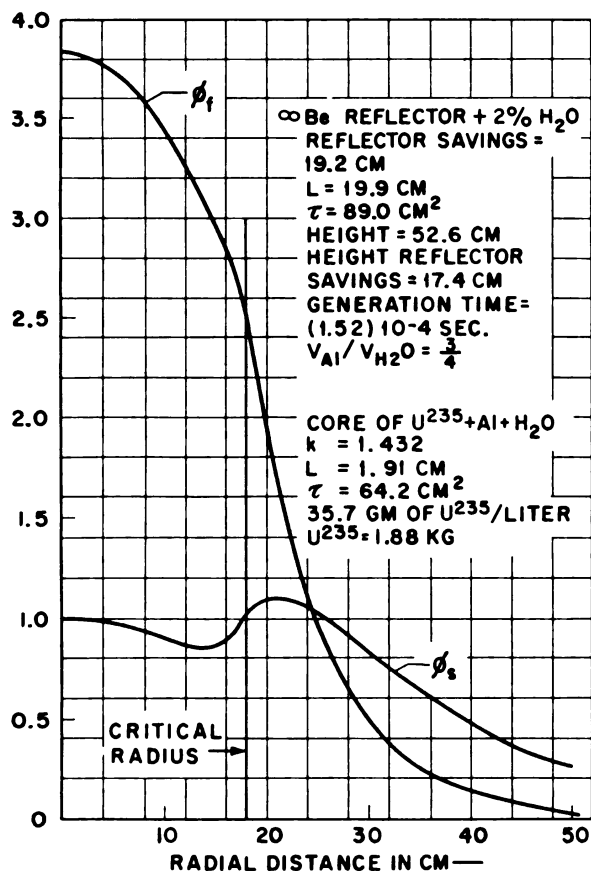


Fig. 3-32 Neutron-flux distributions.

(thermal flux) in the reflector. For a thin-slab reactor the rise of ϕ , in the reflector is most pronounced because of the relatively high rate of thermalization in the beryllium near the core. For this reason a slab-type loading has been used almost exclusively in the MTR. The change from a circular to a rectangular horizontal cross-section reactor hardly affects the neutron-flux distributions (compare Figs. 3-31 and 3-35).

Roughly, the calculations predicted that, if the thermal flux is normalized to 2×10^{14} at the edge of the reactor, it would rise to about 2.3×10^{14}

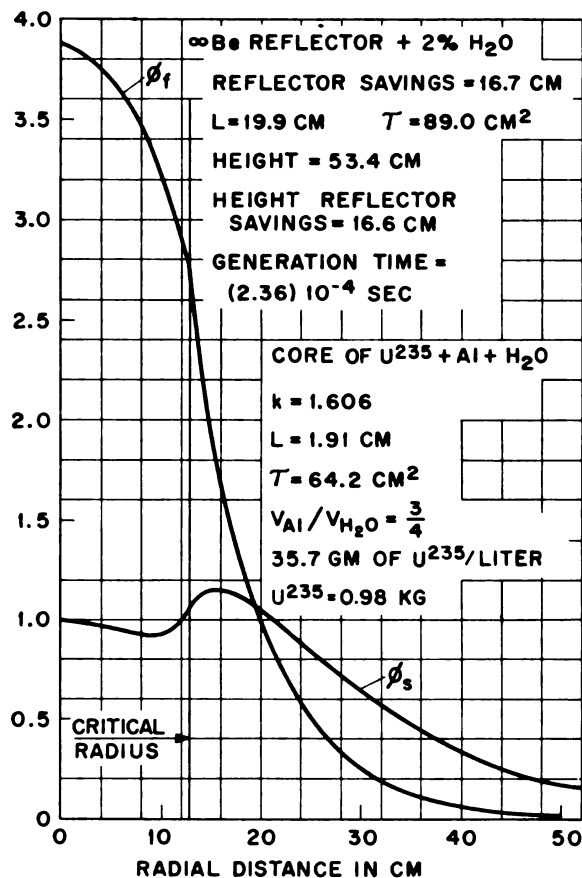


Fig. 3-33 Neutron-flux distributions.

5×10^{14} are being obtained in the reflector (with core control rods partially inserted). The estimate of 6×10^{13} at the tank wall proved to be essentially correct as an average value. However, because of the asymmetry of the loading the actual tank-wall flux varies from about 10^{13} to nearly 10^{14} .

Longitudinally, the design calculations indicated that the thermal flux in the $k = 1.373$ cylindrical reactor (Fig. 3-34) would drop only by a factor of 2 before again rising in the top (or bottom) water-plus-aluminum reflector. Flux

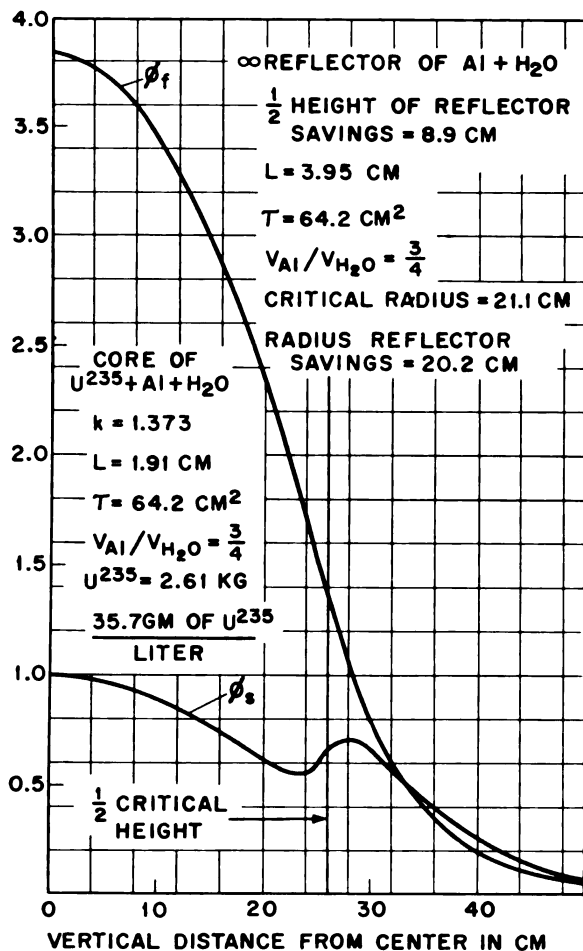


Fig. 3-34 Neutron-flux distributions.

measurements in the MTR later showed this factor of 2 to be essentially correct for most of the fuel assemblies. The flux in the corner assemblies tends to be somewhat flatter.

3-38 Reactivity Losses. For various reasons, such as Xe^{135} and Sm^{149} poisoning, depletion of U^{235} , and temperature rise, the reactivity of the reactor is lower during high-power operation than during zero-power operation. The reactor therefore must be built considerably larger than is necessary, to be critical in the cold, clean state.

With the thermal flux as high as it is in the MTR, the amount of Xe^{135} and Sm^{149} is saturated with respect to increased flux. The reactivity effect $\Delta k/k$ of these poisons at equilibrium depends for high flux only on the relative amount

of fissionable material to diluent. It was estimated that the reactivity effect of the equilibrium Xe^{135} in the MTR would be 3.9 per cent and the Sm^{149} , 1.0 per cent. There has been no experimental evidence to indicate that these figures are much in error.

After shutdown the Xe^{135} produced by I^{135} decay builds up and goes through a maximum at about 11 hr; at this time (11 hr) it was estimated that the additional reactivity loss due to Xe^{135} growth equaled 40.6 per cent. The total extra Xe^{135} loss is less than that which comes from shutdown iodine because any Xe^{135} which is already present at shutdown simply decays without going through a maximum. The time course cal-

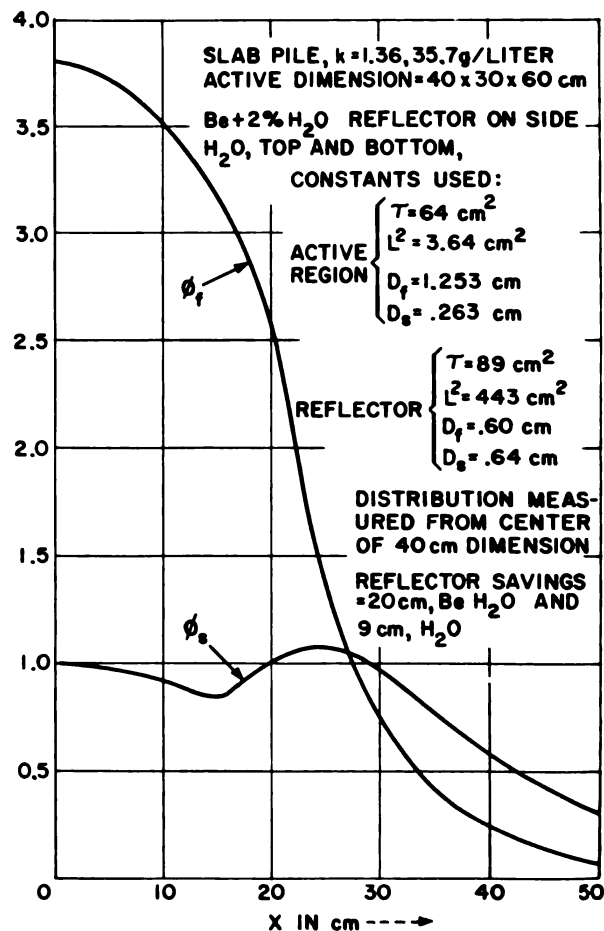


Fig. 3-35 Neutron-flux distributions.

culated for the Xe^{135} which grows from I^{135} is shown in Fig. 3-36 (curve labeled $\text{I}^{135} \rightarrow \text{Xe}^{135} \rightarrow \text{Cs}^{135}$), and the calculated time course of the

Xe^{135} which decays directly is given in the curve labeled $\text{Xe}^{135} \rightarrow \text{Cs}^{135}$.

The Sm^{149} is a daughter of I^{149} ; its concentration therefore increases after shutdown. It was computed that the extra loss due to it at very long times after shutdown would be $\Delta k/k = 3.3$ per cent.

from experience with necessary waiting times until restart becomes possible, it appears that the calculated predictions were reasonable.

In regard to depletion, the theoretical estimates were that the reactor would lose 0.35 per cent $\Delta k/k$ per day at 30 megawatts because of fuel burn-up and accumulation of low cross-section

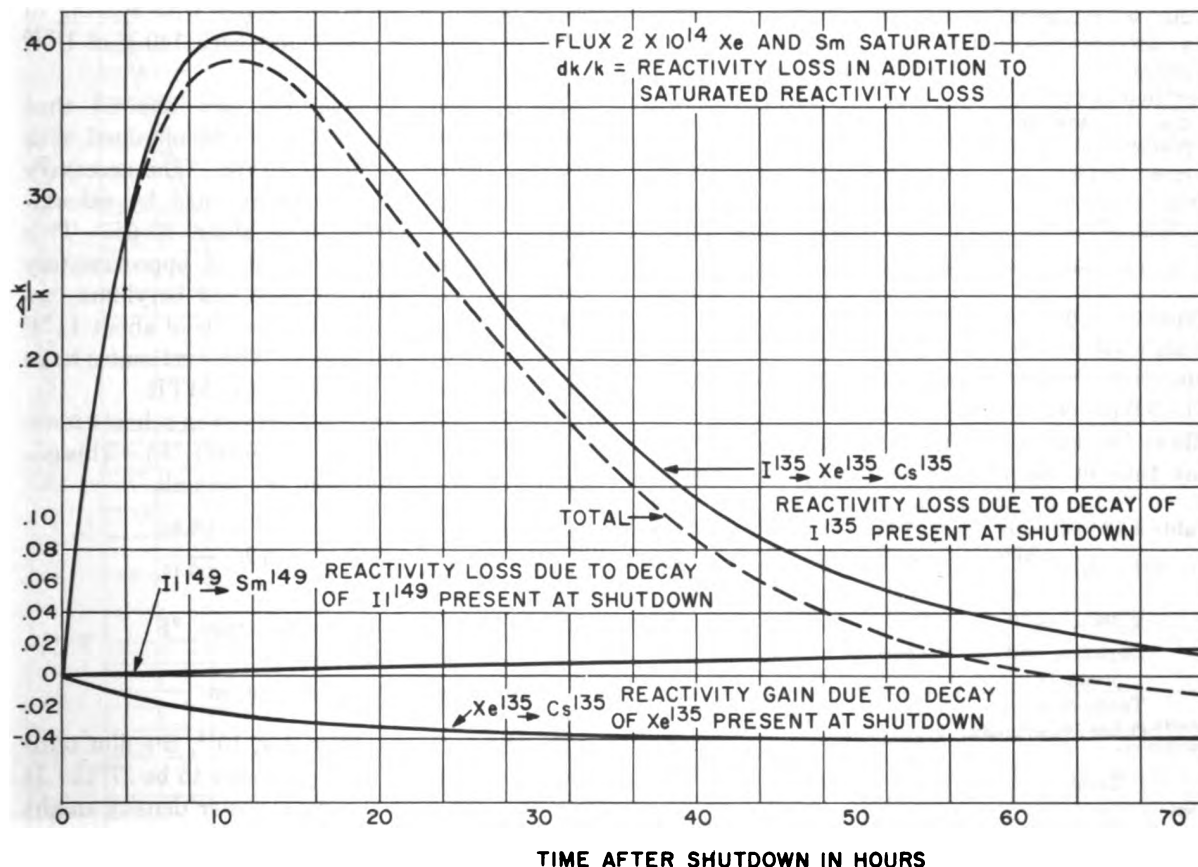


Fig. 3-36 Extra reactivity change after shutdown.

The over-all time behavior that was estimated for the extra reactivity loss is also given in Fig. 3-36. It was seen that, in order to override the $\text{Xe}^{135} + \text{Sm}^{135}$ loss at all times, the calculations indicated that an excess k equal to the sum of the net extra loss, 38.8 per cent, plus the equilibrium loss, 4.9 per cent, or $\Delta k/k = 43.7$ per cent, would be needed.

Since the MTR has been designed with only about 20 per cent excess reactivity (even without experiments) it is not possible to verify the preceding computations experimentally. However,

fission-product poisons. This estimate was rather seriously in error primarily because of the assumption made that the burn-up would be spatially uniform. The correct value proved to be about 0.5 per cent $\Delta k/k$ per day at 30 megawatts.

With regard to temperature coefficient, the calculations are summarized in Table 3-12. These predictions agree well with experimental determinations of temperature coefficients that were made later during the MTR startup exercises.

Table 3-13 gives the breakdown and total excess reactivity that it was estimated the MTR would

Table 3-12 Partial Temperature Coefficient at 20°C, Al-H₂O Neutron Temperature

Reactor (cylindrical radius)	$\begin{cases} r = 17.85 \\ k = 1.432 \end{cases}$	$\begin{cases} r = 21.12 \text{ cm} \\ k = 1.373 \end{cases}$
Coefficient ($\Delta k/k/^\circ\text{C}$) due to expansion of the fuel-bearing Al plates.	-1.4×10^{-5}	-1.3×10^{-5}
Coefficient ($\Delta k/k/^\circ\text{C}$) due to expansion of water between fuel plates	-3.0×10^{-5}	-2.8×10^{-5}
Coefficient ($\Delta k/k/^\circ\text{C}$) due to change in (effective) microscopic cross sections	-10.5×10^{-5}	-8.9×10^{-5}
Total coefficient ($\Delta k/k/^\circ\text{C}$)	-15×10^{-5}	-13×10^{-5}

require. This has proved to be not much less than what is actually required although the consumption is distributed somewhat differently. The 9.6 per cent for Xe¹³⁵ and Sm¹⁴⁹ in Table 3-13 allows for restart up to 30 min after shutdown at any time during the run. As actually operated,

Table 3-13 Minimum Reactivity Losses Which Must Be Allowed for in Reactor

Source of loss	$\Delta k/k, \%$
Xe ¹³⁵ + Sm ¹⁴⁹	9.6
Depletion and low cross-section fission products	3.5
Temperature	0.8
Other (experiments, etc.)	5.0
Total	18.9

the MTR does not have this "grace" period during the later part of a run. If accidental shutdown occurs late in a run, the reactor is simply reloaded. Thus only 4.9 per cent reactivity need be allowed for Xe¹³⁵ and Sm¹⁴⁹, leaving 4.7 per cent here available for depletion. The 0.8 per cent allowed for temperature is somewhat high. The 5 per cent for experiments has proved to be just about right. Since the running time has been 3 weeks per cycle instead of 10 days and since the $\Delta k/k$ loss per day is more than predicted, the 3.5 per cent figure in Table 3-13 is much too low. The result is that the total reactivity required is a few per cent (reactivity) more than the 18.9 per cent of Table 3-13, even using part of the Xe¹³⁵ and temperature allowance for depletion.

HEAT PRODUCTION, TRANSFER, AND REMOVAL

3-39 The nuclear ratio of U²³⁵ to H₂O in the MTR was established from a critical-mass point of view. The thickness of fuel plates and alloy composition were determined from considerations of strength of materials. These considerations led to choice of the 60-mil plates with spacing of 0.117 in. between plates and with 140 g of U²³⁵ per fuel assembly.

Heat-transfer calculations then showed that satisfactory heat transfer could be obtained with a water velocity of about 30 fps. The necessary pressure drop was then determined by calculations and experiments to be about 40 psi. This pressure drop produces a flow of approximately 20,000 gpm through the core and beryllium, resulting in a water temperature rise of about 11°F for 30-megawatt operation. These estimates have all proved roughly correct in the MTR.

The foregoing conditions result in a heat-transfer coefficient of 6300 Btu/(hr)(ft²)(°F). This coefficient is calculated from the formula

$$h = 120(1 + 0.012t) \frac{V^{0.84}}{D^{0.16}}$$

where t = bulk-water temperature, °F

V = water velocity, fps

D = equivalent diameter, in.

With the average flux of 2×10^{14} , the film temperature drop was then predicted to be 27°C. It was thought that the peak power density might be as much as 1.87 times the average, so that the greatest film drop might be as much as 50°C, but this drop would still provide safe fuel-plate temperatures.

The results of the calculations on the heat production in the reflector are plotted in Fig. 3-37. The greatest heat load outside the core occurs in the mid-plane of the beryllium reflector at the interface between beryllium and core. It was calculated that this maximum might be as much as 3.84 cal/(cm³)(sec). If the beryllium were shaped like a slab and cooled on both front and back faces, the temperature rise in the center would be

$$\Delta T = \frac{qx^2}{2k} = 4.8x^2$$

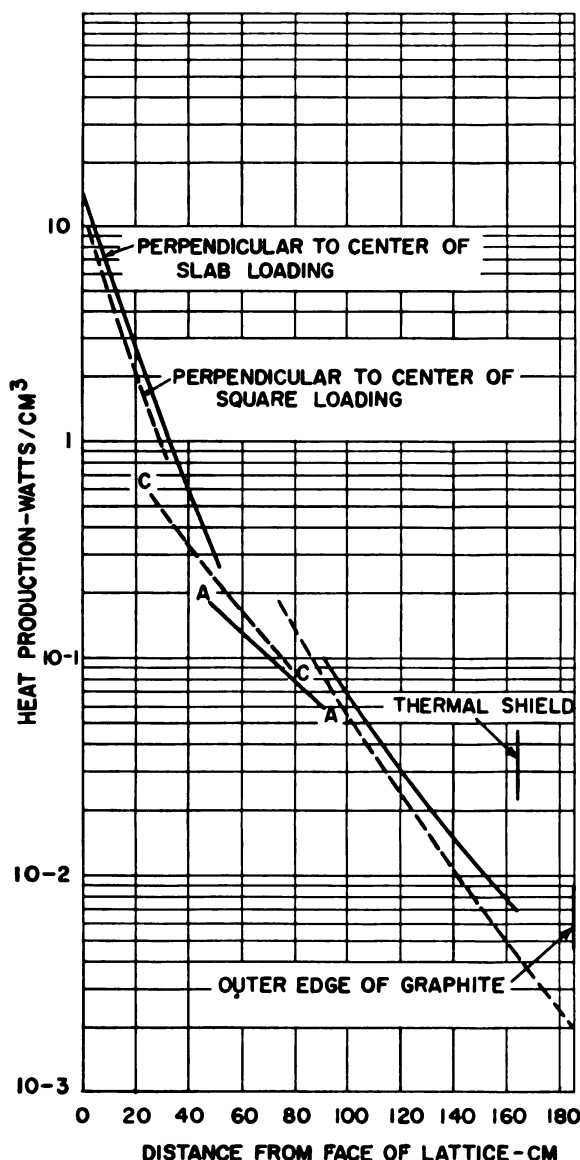


Fig. 3-37 Heat production in reflector for slab or square loading—modified for pebble bed and thermal shield, 30,000-kw operation.

where q is the heat production ($= 3.84$), x is the half thickness of the slab, k is the thermal conductivity and is $0.4 \text{ cal}/(^{\circ}\text{C})(\text{cm})(\text{sec})$. On this basis, and considering a temperature rise of 75°C to be reasonable, it was calculated that 8-cm beryllium slabs would be allowable. Actually the beryllium cooling is accomplished by water flow through small cylindrical holes running vertically through the pieces.

The estimated heating in the graphite pebble zone is shown in the AA section of Fig. 3-37. It was calculated from this that the total heat from the pebble bed (which has been air cooled) should be about 420 kw, with the reactor at 45 megawatts. Similarly it was estimated that the total heating in the permanent graphite would be about 330 kw with the reactor at 45 megawatts.

The heating in the thermal shield was calculated to be 47.5 kw with the reactor at 30 megawatts. This figure assumed that the reactor and reflector gamma rays would produce an amount of heat equal to that produced by capture gamma rays in the steel.

Cooling requirements after shutdown were computed from after-shutdown power levels as obtained from the Way-Wigner formula:

$$P = BF[t^{-0.2} - (t + T)^{-0.2}]$$

where P = Mev/sec

B = constant

F = fission per second per kw

t = seconds after shutdown

T = seconds of reactor operation

Table 3-14 gives the fall off of heat rates and necessary water rates.

Table 3-14

Time	Reactor lattice			Beryllium		Reactor lattice, Be controlling	
	kw	gpm	fps	kw	gpm	gpm	fps
0 sec.....	1950	850	1.95	350	518	3570	8.2
10 sec.....	1120	488	1.12	201	298	2050	4.7
1 hr.....	269	118	0.27	48	72	498	1.13
10 hr.....	129	56	0.09	23	34	236	0.54
4 days.....	45	20	0.05	8	12	82	0.19

Shielding

3-40 In the first discussions of the reactor shield the designers assumed that when gamma-ray radiation predominated, the radiation intensity should be 0.01 r per 8 hr. The permissible neutron dose measured in roentgens equivalent physical (rep) is less than the gamma-ray tolerance in roentgens because of the greater damage ascribed to neutrons.

For use in calculations it is desirable to express these limiting intensities as fluxes. The flux values chosen were as follows:

6 to 10 Mev neutrons.	5 neutrons/cm ² /sec, equivalent to 0.1 mrep/hr
6 Mev gamma rays.	100 photons/cm ² /sec, equivalent to 1.2 mr/hr

These represent the allowed radiation leakage where one of the components is completely dominant. When both neutrons and gamma rays are present, it is necessary to consider the combined effect.

In order to keep the shield from getting too bulky and complicated, it was decided to use barytes ore as the aggregate. This results in concrete with a density of 3.5 g/cm³. The radiation-attenuation characteristics of barytes concrete had not been investigated extensively at the time of the MTR design. Gamma-ray-attenuation experiments using a 6-curie radiocobalt source gave results which showed that the gamma-ray attenuation of barytes concrete as compared with ordinary concrete varies as the ratio of the densities. From experience it was known that a fairly reliable estimate of the fast-neutron relaxation length for a material may be obtained by comparing its total macroscopic cross section with that of a similar material, e.g., concrete of somewhat similar composition, whose asymptotic relaxation length is known. From this method a fast-neutron relaxation length of 8.2 cm was determined, and this was used in the analysis.

The attenuation of gammas was assumed to follow the usual formula

$$\phi_E \approx \phi_0 B e^{-b}$$

where ϕ_E is energy flux of radiation measured in Mev/cm²/sec, b is the thickness of the shield measured in relaxation lengths where the relaxation length is computed from the total absorption coefficient (including Compton scattering), and B is the build-up factor. A conservative approximation is $B = b$.

Since the usual design value for the asymptotic gamma-ray relaxation length in ordinary concrete is 15 cm, the relaxation length in barytes concrete was taken as 10 cm on the basis of inverse proportionality to mass density.

In checking a shield design it is necessary to treat separately the gamma rays and the neutrons which are incident on the inner face of the concrete. The solution of the gamma-ray problem was achieved in the following manner:

The heat production in the thermal shield due to (1) the absorption of capture gamma rays formed in the iron and in the graphite and (2) the absorption of gamma rays from the core had been studied in detail. These reports gave a value of 6.7×10^{-16} watt/cm³ produced at the outer face of the iron per unit thermal-neutron flux at the inner face. It was believed that less than half of this would be due to gamma rays from the core. Since the absorption coefficient $\mu = 0.235$ cm⁻¹ for 8 Mev gamma rays in the iron, the gamma-ray flux at the outer face of the thermal shield is

$$\begin{aligned} \phi_E(O) &= \frac{nv \times 6.7 \times 10^{-16}}{1.6 \times 10^{-13}} \times \frac{1}{0.235} \\ &= 0.018nv \text{ (Mev/cm}^2\text{/sec)} \end{aligned}$$

This states then that at any point on the outside of the thermal shield the gamma-ray energy flux is just 0.018 times the neutron flux at the corresponding point on the inside face of the thermal shield. Since detailed thermal-flux maps had already been prepared, this gave the incident gamma-ray flux for the shield studies. A value of $nv = 10^{11}$ was employed in design as a conservative figure.

Although the neutron-attenuation problem is much more difficult to solve than the corresponding gamma-ray problem, it was possible to adopt a "recipe" to give conservative results. In addition to the neutrons it is also, of course, necessary to worry about the capture gammas which the neutrons generate.

In the recipe adopted for treatment of the fast-neutron attenuation, it was assumed that all fast neutrons entering the concrete would be captured after penetrating the concrete a distance equal to the "age displacement." The secondary gammas so formed are then postulated to all flow outward with the estimated 10 cm relaxation length.

The thermal neutrons and hence captures in the shield were taken to have the distribution

$$\phi_{th}(Z) \approx e^{-\frac{(Z-r/\lambda)}{\lambda}}$$

Thus, the thermal neutrons are distributed exponentially, but they are displaced to a larger value of Z by the distance τ/λ , and this is called the *age displacement*. Values of $\tau = 350 \text{ cm}^2$ and $\lambda = 11 \text{ cm}$ were used, giving an age displacement of about 38.1 cm.

No clear-cut method for estimating the neutron current leaving the thermal shield was established analytically. The final analysis of the concrete with respect to neutron effects was held in abeyance until experimental data were available from the mock-up. Direct measurement of the fast-neutron current was not made, but it was estimated by measuring the thermal flux and cadmium ratios and using the following reasoning:

1. The cadmium ratios of the outer surface of the graphite are very insensitive to changes in distance from the core. This is interpreted to indicate that the spectrum at all positions on the outside of the graphite is nearly at equilibrium.

2. With these data it seems reasonable to postulate that (a) the observed thermal flux at the surface of the graphite is proportional to the total flux there and (b) the cadmium ratios at this location are proportional to the fraction of fast neutrons in the spectrum. This at least would seem to be a conservative procedure.

Table 3-15 presents the results of the thickness calculations. It is interesting that the attenuation

requirements of incident gamma rays are completely dominant.

In addition to the most important problem of determining the adequacy of the concrete portion of the biological shield, there are a number of other shielding problems relative to the design of the MTR associated facilities. Only two will be discussed here because they cover the essential data and methods: (1) shielding requirements of the top plug and (2) shielding of the exit water lines.

The shielding of radiation above the active lattice is accomplished by the water in the reactor tank plus the top plug. The depth of water in the reactor tank is adjusted to provide adequate protection after shutdown of the reactor and after the top plug has been removed. The design established the requirements to be $17\frac{1}{2}$ ft of water plus a 21-in. iron plug. The plug is hollow and is filled to 8 in. with lead shot. The following paragraphs summarize the calculations.

It is assumed that on the average a portion of the core 72 by 24 by 60 cm will contain active fuel elements. With such a loading, there will be about 3.5 kg of U^{235} in the core and at 30 megawatts the average neutron flux will be 2×10^{14} . The gamma-ray source in the active section is given in Table 3-16. The source strength is given by the product of the neutron flux, cross section,

Table 3-15 Required Biological Shield Thickness

Position (1)	Geometrical factor (2)	$n\gamma$ calculated, $n/\text{cm}^2/\text{sec}$ (3)	$n\gamma$ experimental, $n/\text{cm}^2/\text{sec}$ (4)	$n\gamma$ design, $n/\text{cm}^2/\text{sec}$ (5)	Gamma-ray flux emerging from thermal shield, gamma rays/ cm^2/sec (6)	Cd ratio for design (7)	Neutron leakage current emerging from thermal shield, $n/\text{cm}^2/\text{sec}$ (8)	Shield thickness beyond 1 in. to attenuate concrete capture gamma rays, in. (9)	Total shield thickness to attenuate concrete capture gamma rays, in. (10)	Total shield thickness required to attenuate gamma rays emerging from thermal shield, in. (11)	Total shield thickness required to attenuate neutron leakage current, in. (12)
A. Center of N-S vertical faces.	1	4×10^{11}	3×10^{11}	4×10^{11}	2.4×10^8	900	5.8×10^4	44	46	67.2	45
B. Center of top horizontal edges.	$\frac{1}{2}$	0.5×10^{11}	0.5×10^{11}	1×10^{11}	0.6×10^8	1000	1.3×10^4	37.2	49.2	59	40.5
C. Top corners.	$\frac{1}{4}$	0.25×10^{11}	0.1×10^{11}	0.5×10^{11}	0.3×10^8	1000	0.65×10^4	35	47	54	38
D. Center of E-W vertical faces	1	2×10^{11}	2.5×10^{11}	2×10^{11}	1.2×10^8	1000	2.6×10^4	40.5	52.5	64	42.5
E. Vertical edges at center plane.	$\frac{1}{2}$	0.5×10^{11}	0.5×10^{11}	1×10^{11}	0.6×10^8	1000	1.3×10^4	37.2	49.2	59	40.5
F. Top face.	1	2.5×10^{11}	0.7×10^{11}	2×10^{11}	1.2×10^8	1000	2.6×10^4	40.5	52.5	65	42.5
G. Bottom face.	1	6×10^{11}	2×10^{11}	3×10^{11}	1.8×10^8	300	3.9×10^4	41.5	53.5	66	43.5

and gamma-ray energy per capture divided by the core volume. Combining gamma rays of similar energies gives the source spectrum shown

Table 3-16

Source	Reactor cross section $N\sigma$, cm^2	Gamma rays per capture
U fission.....	4900	5 1-Mev fission gamma rays
U capture.....	800	2 2.5-Mev fission-product gamma rays
H ₂ O capture....	1320	1 6-Mev gamma ray
Al capture.....	588	1 8-Mev gamma ray
Th capture....	610	1 6-Mev gamma ray

in Table 3-17. At large distances from the reactor center line, the radiation intensity may be calculated by simple conventional formulas. Thus for

Table 3-17

Gamma source, Mev	Q_v , Mev/cm ² /sec	Watts/cm ² /sec	μ_s , cm ⁻¹
1	4.75×10^{12}	7.6	0.099
2.5	52.6×10^{12}	8.4	0.072
6	16.3×10^{12}	2.6	0.044
8	9.0×10^{12}	1.44	0.039

each of the gamma-ray sources the intensity is given by

$$I = 1.9 \times 10^{-3} \frac{Q_v A}{4\pi\mu_s a^2} e^{-\mu_s a/1.2}$$

where I = intensity, mr/hr

1.9×10^{-3} = conversion factor, Mev to mr/hr

Q_v = source strength, Mev/cm²/sec

A = area of top of active section, cm²

μ_s = absorption coefficient of core, cm⁻¹

a = distance above active lattice = 565 cm

μ = absorption coefficient of water, cm⁻¹

1.2 = factor for forward scattering

Therefore,

$$I = 8.2 \times 10^{-7} \frac{Q_v}{\mu_s} e^{-565(\mu/1.2)}$$

Using this equation and values of Q_v shown in Table 3-17 the resultant gamma-ray intensities I at the top plug are given in Table 3-18.

Table 3-18

Gamma source, Mev	$\mu/1.2$, cm ⁻¹	$564\mu/1.2$	$e^{-\mu a/1.2}$	I , mr/hr
1	0.058	32.8	10^{-14}	
2.5	0.033	18.7	10^{-8}	
6	0.023	13.0	2.2×10^{-5}	670
8	0.020	11.3	1.2×10^{-5}	2300
Total intensity without top plug.....				3000

It is also necessary to verify that adequate water depth would be available after shutdown with the top plug removed.

The total gamma-ray activity following infinite operation of a reactor is given by

$$(Q_0 E) = 6.3ft^{-0.2} \text{ Mev/sec}$$

where f = fissions per second in source

t = time after shutdown

The hard-gamma-ray activity (about 3 Mev) after shutdown, however, depends on the yield and decay of specific fission-product chains. These have been measured by the photoneutron threshold reaction in heavy water. The average lifetime and yield of the gamma rays of energies higher than this value are given in Table 3-19.

Table 3-19

t (mean)	A_f (yield per fission)	E (energy), Mev
76 hr.....	0.0015	~3
6.3 hr.....	0.0042	~3
2.4 hr.....	0.045	2.6
39 min....	0.038	2.6
11 min....	0.046	3
3.4 min....	0.132	2.6
59 sec.....	1.58	2.25
3.6 sec....	0.675	3.5

Using the same method as in the calculation of the radiation intensity during operation of the reactor, but considering only the hard fission-product gamma-rays given in Table 3-19, the gamma-ray intensity at the top of the reactor well may be calculated as a function of the time after shutdown. In this case the source term at 30 megawatts power is given by

$$30,000 \times 3.1 \times 10^{13} = 9.3 \times 10^{17} \quad \text{fissions/sec}$$

$$Q_v = \frac{(9.3 \times 10^{17})(A_\gamma E)_i}{24 \times 72 \times 60} \quad \text{Mev/cm}^3/\text{sec}$$

and

$$Q_v = 9 \times 10^{12}(A_\gamma E)_i$$

The radiation intensity at the surface of the water after shutdown is

$$I = 6 \times 10^6 (A_\gamma E)_i \frac{e^{-530(0.035)t}}{\mu_s}$$

Assuming average energy of hard gamma rays to be 2.75 Mev,

$$I = 1.4(A_\gamma E)_i \quad \text{mr/hr}$$

Values of $(A_\gamma E)_i$, Q_v , and the gamma-ray intensity at various times after shutdown are shown in Table 3-20. This indicated that the water would

Table 3-20

Time after shutdown	$(A_\gamma E)_i$, Mev/fission	Q_v , Mev/cm ³ /sec	I , mr/hr
10 sec.....	4.37	39.5×10^{12}	6.2
10 min.....	0.32	2.9×10^{12}	0.5
1 hr.....	0.11	1.0×10^{12}	0.15
1 day.....	0.003	0.027×10^{12}	0.04

give adequate protection following an elapsed time greater than 10 min after shutdown of the reactor.

The final shielding problem to be discussed is the shielding of the exit water lines. The activity of the water leaving the reactor is due mainly to the N^{16} activity formed from the reaction $O^{16}(n,p)N^{16}$. This is a fast-neutron reaction having a threshold at about 9.5 Mev and a cross section of about 1 barn. Since a small fraction

(about 1/50,000) of the fission neutrons have energies above 9.5 Mev, the effective cross section per fission neutron is about 0.014 mb, or 1.4×10^{-29} cm². The activity of the water leaving the reactor is given by

$$A = nv\sigma_f(1 - e^{-\lambda T})e^{-\lambda t}$$

where nv = fast flux in lattice = 10^{14} neutrons/cm²/sec at 30 megawatts

σ_f = cross section for N^{16} production = 0.014 mb or 4.68×10^{-7} cm² per cubic centimeter of water

λ = N^{16} decay constant = 0.092 sec⁻¹

T = exposure time in lattice = $\frac{1}{2} \times 5$ sec

t = decay time after leaving lattice

A = activity, dis/cm³/sec

Thus

$$A = 2.88 \times 10^5 e^{-\lambda t} \quad \text{dis/cm}^3/\text{sec}$$

Since the average energy of the N^{16} gamma ray is 6.5 Mev, the water activity is $1.85 \times 10^6 e^{-\lambda t}$ Mev/cm³/sec.

The required shielding of the exit water lines varies with the distance from the active core owing to the relatively rapid decay of N^{16} . The activity at critical locations along the exit water lines was calculated from the flow rate and corresponding decay times. The water activity obtained at strategic points is summarized in Table 3-21, based on a flow rate of 20,000 gpm.

Table 3-21

Location	Decay time, sec	Water activity from N^{16} , Mev/cm ³ /sec
A. Leaving reactor tank.....	2.3	1.5×10^6
B. Vertical rise at reactor center line.....	4.1	1.28×10^6
C. Center of top of loop.....	6.6	1.0×10^6
D. Leaving reactor structure.....	15	4.6×10^5
E. Leaving reactor building.....	26	1.7×10^5
F. Entering seal tank.....	63	5.5×10^4

The necessary shielding was then calculated in a standard fashion.

PART 3 PUTTING THE MTR IN OPERATION

PREOPERATIONAL TEST PROCEDURES

3-41 In preparation for the startup of the MTR, the operator prepared a manual outlining the tests considered necessary to demonstrate the operability of all components of the reactor and to establish the principal reactor constants, such as critical mass, excess reactivity, and maximum operating power level.

This manual was divided into two sections. Part I was called Pre-Neutron Tests and covered the testing of the entire plant, as well as hydraulic, dimensional, instrumental, and mechanical tests of the reactor proper. Part II, Post-Neutron Tests, with which this part is concerned, covered the experiments performed from the time that the first neutrons were generated in the reactor to the end of the startup program.

In the main, this manual and its proposed schedule were followed. When it became apparent, following the first complete cycle of burn-up of a full loading, that the operating cycle was shorter than had been anticipated, additional experiments were performed to determine the factors responsible for the discrepancy.

To give an idea of the scope of the tests scheduled in Part II, excerpts are given below:

I. Approach to Criticality. Prior to this, all pre-neutron tests are to be completed. Additional instrumentation will be inserted for the purpose of the tests.

II. Characteristics of Reactor

A. Check of Control System. At 10 to 100 watts the control system and instruments will be checked using reactor-derived signals.

B. Shielding Survey. The shielding will be checked as power levels are increased.

C. Attainment of Full Slab Loading. The control rod positions as a function of the fuel loading will be followed.

D. Neutron Checks on All Instruments. Operating at 1000 watts, all the chambers will be given a more extensive check.

E. Map of Neutron Distribution. The neutron flux and flux distribution will be determined by

using foils, chambers and cobalt wires. The power level will be about 1 kw.

F. Calibration of Control Rods. At about 300 watts a calibration of the regulating rods will be performed.

1. Period Calibration of Regulating Rods. The reactor will be brought from 1 watt to 300 watts on a 60 second period by use of the regulating rod and at various insertion positions of the rod. These experiments will be repeated with different shim rod configurations and combinations to interrelate the shim and regulating rods.

2. Poison Calibration of Control Rods. By inserting stainless steel poison strips uniformly in the lattice, the control rods will be calibrated throughout their length. This will require several separate insertions and measurements.

G. Determination of the Temperature Coefficient. The over-all temperature coefficient will be determined by operating at about 100 watts using externally heated water to vary the temperature of the reactor.

H. Servo-mechanism Operation. At the lowest power at which the servo system will operate (say 10,000 watts), a check of its performance will be made.

I. Power Calibration of Instruments. At a power level of 1-5 megawatts, a heat balance will be made on the reactor and the readings of the neutron and power measuring instruments will be correlated.

J. Other Power Checks

1. Monitoring Instruments. At this power level (1-5 megawatts) the instrumentation of the monitoring tubes under the fuel assemblies and the main N-16 activity monitors will be checked.

2. Shielding. A thorough check of the activity at the reactor structure faces will be made.

3. Temperature Map. The initial temperature map of the reactor structure will be made at these powers.

4. Gamma Heating. A series of measurements will be made to determine the magnitude and distribution of gamma heating in the reactor tank.

K. Determination of Poisoning Effects. The reactor will be operated at 1-5 megawatts long enough

to establish approximate equilibrium in poison buildup and then held at a very low power while the decay curve of the poisons is followed.

III. Establishment of Full Power Rating

A. Intermediate Power Measurements. With the reactor at temperature equilibrium at a power level between 5 and 30 megawatts, power level calibrations and other determinations in II-I and II-J will be repeated.

B. Attainment of Power Rating. Part A will be repeated with stepwise increase in power either until 30 megawatts is reached or until some limiting condition is encountered which forbids further increase.

IV. Full Power Operation. The reactor and its auxiliaries will be operated for 30 days at the maximum power obtained in III-B. During this period the performance of all components will be studied and evaluated. In addition, several major scrams during full power operation, with simulated multiple failures, will be initiated. It will be determined that the reactor and the plant as a whole behave satisfactorily.

The startup program, as it was actually carried out, is described in some detail in the following.

POSTNEUTRON TESTS

Before loading fuel into the reactor, a neutron source and the additional instrumentation which was needed for the approach to criticality were

installed. A polonium-beryllium source with a strength of approximately 5 curies was placed at the east end of the lattice in a tube which runs through the reactor a little below the center line of the fuel. Fission chambers were located in two vertical holes VG-8 and VG-18 (Fig. 3-38) in the graphite reflector at the center line of the reactor. A boron trifluoride counter was placed in the horizontal beam hole HB-4 in a position as near the lattice as possible. After all the appropriate counting equipment was installed and checked, background counts were taken on these chambers. As an added safety measure, all neutron-sensitive chambers which form a part of the reactor control system were inserted in the reactor to the position of greatest sensitivity. All period and level safety instruments were made active.

First Approach to Criticality

3-42 The first group of fuel assemblies was loaded into the reactor, and measurements were made of the equilibrium counting rates of each detector with control rods inserted, partly inserted, and fully withdrawn. The same procedure was followed for six such loadings, at which time the reactor became critical with a period of 100 sec.

In Fig. 3-39 the reciprocal counting rates observed are shown plotted against mass of fuel in the core. The mass values used were taken to be

Table 3-22 First Approach to Criticality

Run number	Total mass of U^{235} , g *	Counts/min, VG-8	Counts/min, VG-18	Counts/min, HB-4
Background (shim rods fully inserted, no fuel)	0	6.7	1.1	118.1
Background (shim rods fully withdrawn, no fuel)	0	9.3	2.6	326
1	666	21.4	5.0	812.2
2	1,149	54	32.5	2,410
3	1,300	78.3	49.2	4,385
4	1,451	150.7	97.9	8,114
5	1,613	587	452	56,384
6	1,775			
Estimated critical mass, g.		1,650	1,650	1,670

Weighted average of critical mass = 16,600 \pm an estimated 15 g.

* Includes fuel in assemblies plus estimated contributions from shim rods.

the sum of the assayed masses of the fuel forming the core and the estimated contribution of fuel in those shim rods which were outside the core. The

determined from Table 3-22 and Fig. 3-40 is 1660 ± 15 g. This is the average of the values obtained from the counters weighted according to

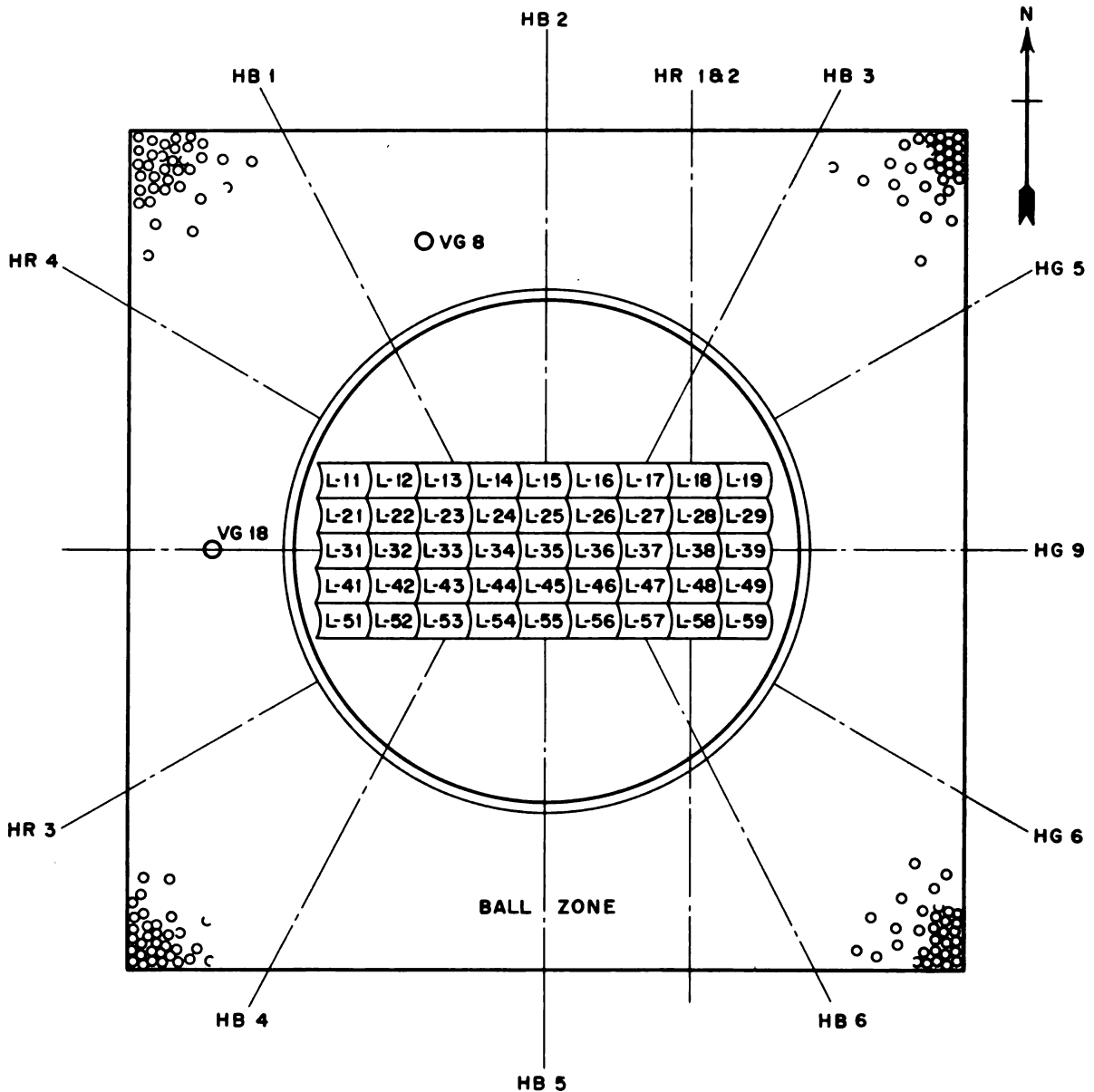


Fig. 3-38 Section through MTR tank and graphite ball section.

extrapolation of these curves to zero ordinate gives the value of the critical mass. A summary of the data pertaining to this critical experiment is given in Table 3-22.

The critical mass for this type of loading as

their probable errors and reliability. The barely critical assembly consisted of 11 fuel assemblies and two fuel-bearing shim rods. A small estimated contribution from the other two fuel-bearing shim rods has been included.

With the first critical assembly, a number of tests on the control system were performed before loading additional fissionable material. Using reactor-derived signals, all neutron-sensitive chambers were checked to see that the current output changed in accordance with neutron-level changes. This was the first point at which there was real assurance that the safety channels were operative.

First Full Loading

3-43 Following the control-system checks with the barely critical assembly, the loading of fuel was continued toward the west end of the lattice until a full 3×9 slab was obtained. At each stage, as may be seen in Fig. 3-40, beryllium pieces were placed in positions adjacent to the fuel at the west end to complete the reflector. The remaining positions were occupied by shim rods and dummy fuel assemblies. At all times during which loading of fuel was taking place, observers were stationed at each of the counters to detect any abnormal changes in counting rates, as a safety measure. The withdrawal procedure for the shim rods was as follows: The beryllium shim rods were first withdrawn completely, and then two of the set of four fuel-bearing shim rods were withdrawn. One of the two remaining rods was taken about two-thirds of the way out, and criticality was reached by moving the remaining rod. The purpose of this scheme was twofold: (1) It provided safety rods in sensitive positions so that the reactivity could be decreased quickly should a short period somehow occur. (2) It was possible to reach criticality with a rod which was relatively insensitive so that criticality could be approached gradually.

When the first 3×9 loading was completed, the reactor contained 3457 g of U^{235} and the fuel-bearing shim rods were at about their mid-positions.

Before operating the reactor at higher powers, it was necessary to make a rough calibration of neutron-measuring instruments in terms of reactor power. For this purpose flux measurements, to be described later, were made at several points in the reactor using indium, gold, and cobalt activation. The instrument constants were estimated by comparing the values so obtained with the design flux maps.

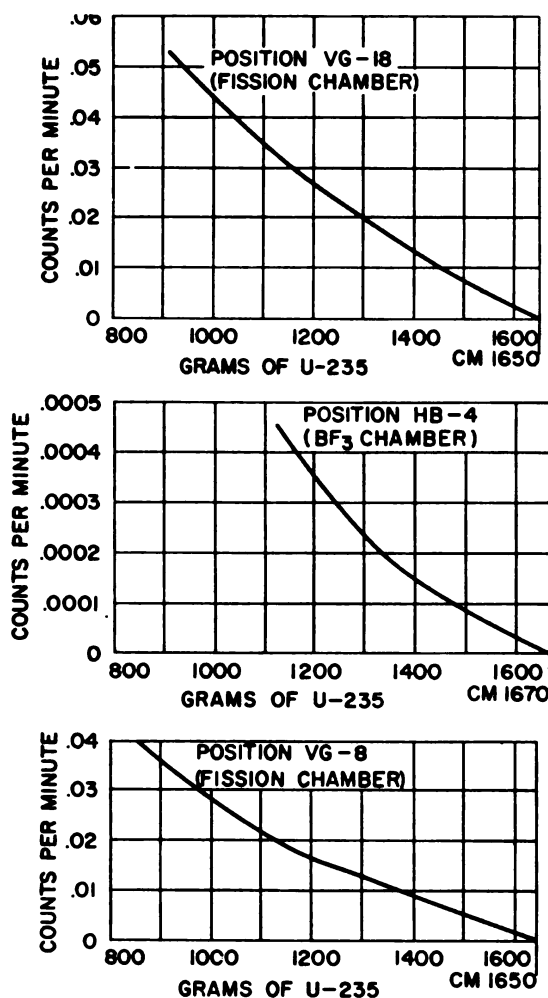


Fig. 3-39 First approach to criticality.

Survey of the MTR Shield

3-44 A complete survey of the MTR shield was made during startup to determine its adequacy from a health-physics point of view. The survey was made with the common detection instruments used by health physics and with x-ray film. Results of this survey may be summarized as follows:

No detectable radiation leaks were observed in the bulk shielding which might have resulted from faulty construction.

The face of the thermal column emitted slow neutrons and gamma radiation in sufficient intensities for each to represent the maximum permissible level of biological tolerance. A $\frac{1}{4}$ -in. sheet of boral almost completely eliminated the slow neutrons.

RUN NO. 1

DF	DF	DF	DF	DF	Be	Be	F ₁	F ₁
DF	⁴ SF	DF	³ SF	DF	² SF	Be	¹ SF	F ₁
DF	DF	DF	DF	DF	DF	Be	Be	F ₁
Be	⁵ OUT	Be	⁶ Be	Be	⁷ S Be	Be	⁸ S Be	Be
Be	Be	Be	Be	Be	Be	Be	Be	Be

RUN NO. 2

DF	DF	DF	DF	Be	Be	F ₂	F ₁	F ₁
DF	⁴ SF	DF	³ SF	Be	² SF	F ₂	¹ SF	F ₁
DF	DF	DF	DF	Be	Be	Be	F ₂	F ₁
Be	⁵ OUT	Be	⁶ S Be	Be	⁷ S Be	Be	⁸ S Be	Be
Be	Be	Be	Be	Be	Be	Be	Be	Be

RUN NO. 3

DF	DF	DF	Be	Be	Be	F ₂	F ₁	F ₁
DF	⁴ SF	DF	³ SF	Be	² SF	F ₂	¹ SF	F ₁
DF	DF	DF	DF	Be	Be	F ₃	F ₂	F ₁
Be	⁵ OUT	Be	⁶ S Be	Be	⁷ S Be	Be	⁸ S Be	Be
Be	Be	Be	Be	Be	Be	Be	Be	Be

RUN NO. 4

DF	DF	DF	Be	Be	F ₄	F ₂	F ₁	F ₁
DF	⁴ SF	DF	³ SF	Be	² SF	F ₂	¹ SF	F ₁
DF	DF	DF	DF	Be	Be	F ₃	F ₂	F ₁
Be	⁵ OUT	Be	⁶ S Be	Be	⁷ S Be	Be	⁸ S Be	Be
Be	Be	Be	Be	Be	Be	Be	Be	Be

RUN NO. 5

DF	DF	DF	Be	Be	F ₄	F ₂	F ₁	F ₁
DF	⁴ SF	Be	³ SF	Be	² SF	F ₂	¹ SF	F ₁
DF	DF	DF	Be	Be	F ₅	F ₃	F ₂	F ₁
Be	⁵ OUT	Be	⁶ S Be	Be	⁷ S Be	Be	⁸ S Be	Be
Be	Be	Be	Be	Be	Be	Be	Be	Be

RUN NO. 6

DF	DF	Be	Be	F ₆	F ₄	F ₂	F ₁	F ₁
DF	⁴ SF	Be	³ SF	Be	² SF	F ₂	¹ SF	F ₁
DF	DF	DF	Be	Be	F ₅	F ₃	F ₂	F ₁
Be	⁵ OUT	Be	⁶ S Be	Be	⁷ S Be	Be	⁸ S Be	Be
Be	Be	Be	Be	Be	Be	Be	Be	Be

Be - BERYLLIUM PIECE

F - FUEL ELEMENTS, SUBSCRIPT REPRESENTS THE

RUN IN WHICH ELEMENT WAS ADDED.

SF - SHIM SAFETY ROD WITH FUEL.

SBe - SHIM SAFETY ROD WITH BERYLLIUM.

DF - DUMMY FUEL ELEMENTS

OUT - POSITION EMPTY

Fig. 3-40 First approach to criticality.

All the principal experimental holes emitted strong beams of gamma radiation which were well collimated and circular, outlining the periphery. The single step in the plugs appeared to be insufficient to prevent streaming. The key slot which looked almost directly at the lattice was particularly intense in neutrons and gamma radiation. Corrective measures consisted of revising the key slot and making special external shielding plugs. These measures have reduced the maximum beam intensity to maximum permissible levels or below.

Small areas of activity were observed near the utility connections which cooled the experimental plugs or experiments. These were either below tolerance or easily shielded. At two locations near the exit air lines, activity was detected where the shield had an indentation. No additional gamma activity, however, was detected opposite the shielded exit water lines.

The level of slow neutrons during full-power operation was readily measured with a BF_3 scaler. Intensities observed coming through the bulk shielding were of the order of 10^{-4} times that of maximum permissible levels. The value increased one or two orders of magnitude in the vicinity of experimental holes; these values were still well below tolerance.

Of interest also was the effectiveness of the subreactor-room shield during the discharge of spent fuel assemblies. Results showed that a person should receive less than the maximum permissible daily dose if he worked outside the subreactor room during the discharge of a complete fuel loading.

Control-rod Calibration

The next important phase of the startup of the MTR was the calibration of the control elements and the shim and regulating rods. Since the experiments which were to follow depended upon these calibrations, a considerable amount of time was spent on them.

3-45 Period Calibration of Regulating Rods.

A vital feature of the safety system of the MTR is the assumption that the total reactivity in the regulating rod is less than the delayed neutron fraction and that consequently instantaneous withdrawal of the rod will not result in prompt

criticality. Therefore, before putting the reactor on automatic control, it was necessary to determine that the total reactivity in the regulating rods was within the proper limits.

The first sensitivity calibration of the regulating rods was done by a period technique. In the MTR the inhour relationship does not hold precisely because of the contribution of the (γ, n) neutrons from the beryllium reflector. The magnitude of this effect was uncertain at the time of startup; therefore the calibration was done in such a way as to be independent of the shape of the inhour curve. In addition, an effort was made to minimize the effects of temperature and power. The measurement was made in the following manner:

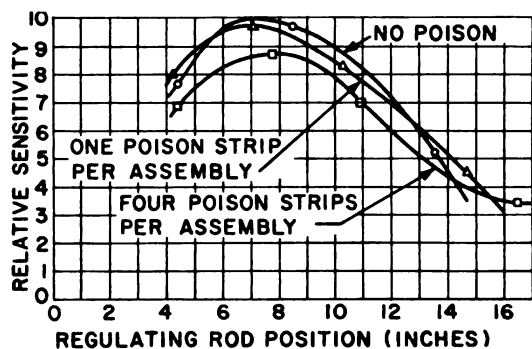
The regulating rod which was to be calibrated was fully inserted and the reactor made critical at the reference power level at which all final measurements were to be made. The four fuel-bearing shim rods were approximately level, and the beryllium-bearing shim rods were withdrawn to their upper-limit positions. After equilibrium had been established, the power level was dropped approximately two decades and maintained constant at this lower value for about 20 min. The regulating rod was then moved manually to put the reactor on a rising period in the neighborhood of 60 sec. The final adjustment to obtain this period was made at least 5 min before the higher-level reference power was reached. This procedure was used in order that the period observed would be free from transient effects. The actual period was read on a period meter with an expanded scale as the reactor power went through the reference value. The reactor power was again leveled out at the higher reference value using a shim rod, leaving the regulating rod in the position obtained in the previous step. The cycle just described was repeated until the regulating rod had been virtually withdrawn from the reactor. The data obtained in these experiments are given in Table 3-23. This method eliminated the effect of the shape of the inhour function on the sensitivity curve by using only one period for all the measurements and, hence, only one point on the inhour curve.

An estimate of the total $\Delta k/k$ was made by assuming the inhour relation to be valid for the MTR and summing the increments of reactivity

Table 3-23 Regulating-rod Calibration by Periods

Point	Poison strips per assembly	Shim-rod position at high power, in.	Shim-rod position at low power, in.	Regulating-rod position, in.	Period by galvanometer, sec.
0	2.69	
1	0	17.589	17.507	6.0	63
2	0	17.331	17.247	8.35	70
3	0	17.075	16.992	11.13	63
4	0	16.821	16.740	15.86	67
5	0	16.557	16.474	21.6	
		16.424	16.603		
0	2.65	
1	1	17.984	17.904	5.86	65
2	0	17.698	17.614	8.49	65
3	0	17.416	17.337	11.48	69
4	0	17.144	17.058	16.76	58
5	0	16.867	16.763		
0	2.66	
1	4	19.098	18.964	6.325	65
2	0	18.747	18.611	9.125	69.2
3	0	18.416	18.286	12.81	
4	0	18.081	17.948	20.120	
		17.760			

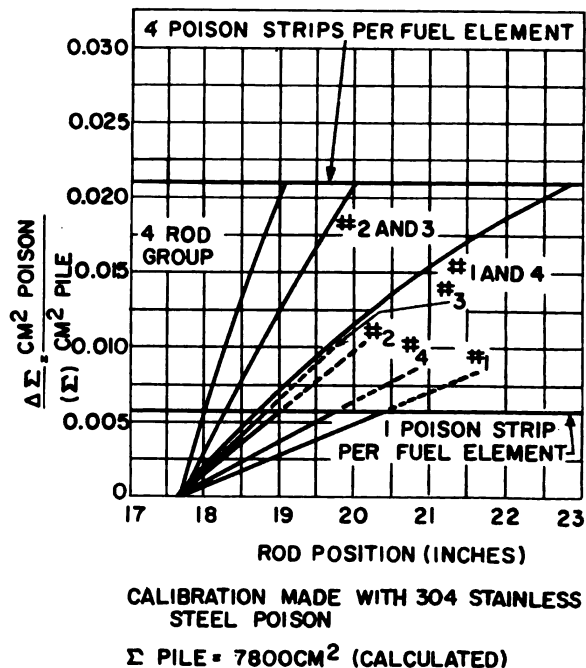
corresponding to the periods obtained at each point on the sensitivity curve. A value of 0.5 per cent for the entire regulating-rod withdrawal was obtained. It may be seen from Fig. 3-41 that the

**Fig. 3-41** Regulating-rod sensitivity curve.

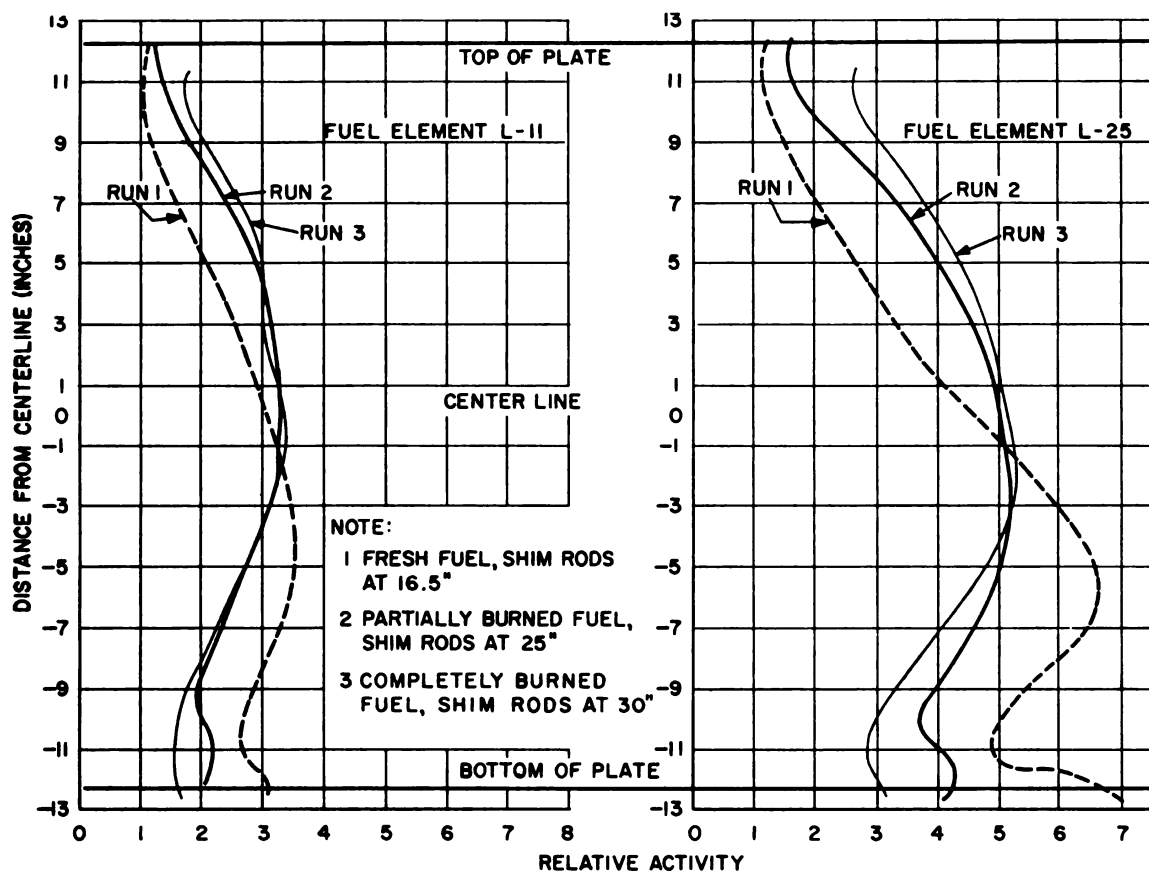
maximum sensitivity of the regulating rod occurs below the mid-plane of the reactor. The reason for this will become apparent in the discussion of the vertical flux distribution in the MTR.

3-46 Calibration of Shim Rods. A preliminary calibration of the shim rods, using stainless-steel poison strips, was made in the first 3×9 loading. These strips were approximately $\frac{1}{4}$ in.

wide, 0.075 in. thick, and 24 in. long with an average weight of 65 g. The material used was 304 stainless steel. An average value of $0.03 \text{ cm}^2/\text{g}$ was taken as the nominal macroscopic cross section at 0.025 ev. The procedure followed was to load an equal number of the stainless-steel strips into each fuel assembly and to determine the critical rod positions. The amount of poison was limited by the space available to approximately 4 per cent $\delta\Sigma/\Sigma_{\text{pile}}$, where Σ is the macroscopic cross section of the poison and Σ_{pile} is the total reactor cross section. Individual and group calibrations were made for four fuel-bearing shim rods, 1, 2, 3, and 4, which have lattice positions 28, 26, 24, and 22, respectively (see Fig. 3-38). No correction was made at this time for the displacement of water by the poison strips or for the effect of changing the constants of the core material. Figure 3-42 is a plot of the calibration obtained in this manner.

**Fig. 3-42** Shim-rod calibration for first 3×9 loading.

Extrapolation of the group calibration curve of the shim rods indicated that the total reactivity available for this loading was approximately 13 per cent. This was the first indication that the available reactivity was less than the design value.

Fig. 3-43 Vertical-flux traverses for first 3×9 loading.

Temperature Coefficient

3-47 Before proceeding to high power levels and the attendant higher reactor temperatures, the temperature coefficient was determined.

The temperature coefficient of the MTR is influenced by the following:

1. Change in absorption, scattering, and fission cross sections of moderator, fuel, and reflector materials because of the change of the average energy of the neutrons.

2. Change in leakage because of density changes.

3. Linear expansion of the control elements.

The indicated reactivity changes (the register readings of the shim-rod indicators) are the corrections which are ordinarily used when experiments are performed at different temperatures. To compute true reactivity losses due to operation at different equilibrium temperatures, the in-

dicated value must be corrected for control-rod expansion.

It would be desirable to know the individual contributions of the beryllium, fuel, and water to the temperature coefficient. However, for the purpose of startup and operation of the MTR it was felt necessary to measure only the composite temperature coefficient. Hence, the following experiment was performed:

Under conditions of constant water flow with the reactor just critical, the bulk temperature of the water flowing through the lattice was raised in steps of approximately 10° from 70 to 90°F . As this was done the resulting motion of the regulating rod was observed at each step. Conversion to $\Delta k/k$ was made on the basis of the poison calibration of the shim rods, which had been performed earlier. The results obtained in this measurement are given in Table 3-24, in which the constants of the control rods were taken from the calibration

curves in Figs. 3-41 and 3-42. A value of $8 \times 10^{-6} \Delta k/k$ per $^{\circ}\text{F}$ was found for the 65 to 90°F range.

Table 3-24 Results of Temperature-coefficient Measurement

Temperature, $^{\circ}\text{F}$		Resulting shim-rod motion, in.		Temp coeff neglecting rod expansion, $\Delta k/k$ per $^{\circ}\text{F}$	Temp coeff deducting rod expansion, $\Delta k/k$ per $^{\circ}\text{F}$
Initial	Final	Shim 2	Shim 3		
64.9	76.6	0.263	11×10^{-6}	9×10^{-6}
76.6	86.6	0.197	9×10^{-6}	7×10^{-6}
86.6	76.9	0.192	9×10^{-6}	7×10^{-6}
76.9	62.1	0.049	0.251	10×10^{-6}	8×10^{-6}
Average.....				10×10^{-6}	8×10^{-6}

It should be remarked that the temperature coefficient determined with the entire reactor tank at the same temperature is not truly representative of the operational value, since the plate temperature varies from top to bottom and the reflector, fuel plates, and moderator are all at different temperatures under the usual operating conditions.

Flux Measurements

3-48 Rough Power Calibration. Before the reactor was operated at levels which were high enough to permit the calorimetric determination of the power, a rough measurement was made using an activation technique.

Gold, cobalt, and indium were used in the form of wires and foils. These materials were placed at the center of HR-2 and irradiated at a low power. The saturated activities were determined by the usual counting-room techniques. The data are given in Table 3-25. The conversion of indium-foil saturated activities to thermal-flux values was made using a factor obtained in standard reactors at other laboratories with foils of the same dimensions. The similarity between MTR counting systems and those previously used was such that this was adequate for the precision required at this time. These values will be converted to ab-

solute values when the MTR standard reactor is completed. Gold measurements were calibrated against a gold foil irradiated at Oak Ridge in a

Table 3-25 Preliminary Power Calibration
(By activation measurements)

Detector	$n\sigma_A^*$	Calculated power †
Gold foils.....	9.4×10^9	
Cobalt wires.....	9×10^9	
Indium foils.....	1×10^{10}	
Average.....	1.0×10^{10}	3 kw

* Thermal fluxes were obtained from measurements with and without cadmium around detectors, except for cobalt, which was done with and without a cadmium-indium shield around the detectors.

† Estimated from the relations between flux and power.

known flux. Cobalt values were based on an estimate of the counting efficiency and a thermal cross section of 34 barns.

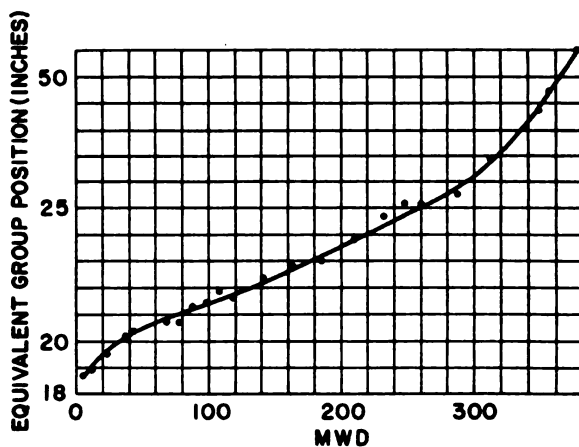


Fig. 3-44 Shim-rod position vs. megawatt-days of operation for 3×9 loading.

3-49 Flux Mapping. Because of the complicated core geometry of the MTR, the space variation of flux, both with regard to magnitude and energy distribution, cannot be calculated directly. Since so many of the computations of reactor behavior and experimental irradiations involve the flux, it was felt desirable to measure in as great detail as possible the flux throughout the entire

reactor. The usual foil techniques were not feasible because of the limited accessibility of certain points in the reactor, such as the fuel assemblies. The time which elapses after a shutdown before a sample can be removed is so long that only materials of low cross section may be used in order that the exposure be large compared to the *not* accumulated from neutron background after shutdown. Cobalt was selected because of the good compromise between the various factors involved. With an activity having a half-life of 5.4 years, a thermal cross section of 34 barns, a simple decay scheme, a fairly energetic radiation, and other favorable physical properties, the activation can be done at sufficiently high power level and for such times that the exposures can be precisely determined.

Considering the structural form of the core and the reflector, the use of strips or wires was clearly indicated. It was finally decided to use small-diameter wires with a retaining ring fashioned on one end from the wire itself. Because cobalt is magnetic, it was possible to remove the wires from the locations in the reactor by means of a magnetic pickup device. The procedure finally adopted consisted of loading as many wires as was consistent with the allowable poison in the reactor, irradiating them for the required time, removing them from the reactor, and scanning the wires with a collimated counter. A scanning device was developed for this purpose which consists of a rack for translating the wires, a collimated proportional counter, and suitable detecting and recording apparatus for handling the data. By this method a point-by-point picture of the activation could be determined.

The flux-mapping program followed during the startup period consisted of making an extensive loading of cobalt wires with each new fuel charge, placing wires in each fuel assembly and in selected beryllium A and L pieces. The beryllium-bearing shim rods were fully withdrawn to the upper-limit positions, thereby completing the beryllium reflector in those positions, and the four fuel-bearing shim rods were level. After irradiation the wires were removed for counting.

As the burn-up progressed and the rods came out, subsequent irradiations were performed in the fuel assemblies and a few beryllium pieces. When the rods had been withdrawn to a point where it

was desired to get a picture of the flux distribution, the reactor was shut down, wires were loaded in, the top plug was replaced, and the shim rods were returned to their positions just before shutdown. After the xenon had decayed sufficiently and the reactor became critical, the power was elevated to the irradiation level.

Most exposures were done at 60 kw for 1 hr. In this manner flux distributions were obtained during several stages of the burnout cycle on 3×9 and 5×5 lattice configurations. In order to evaluate the resonance effect in the cobalt wires, cadmium and cadmium-indium sleeves were placed on some of the wires at several points along their lengths.

The principal use of these vertical flux traverses during startup was the determination of the maximum-to-average flux so that maximum plate temperatures could be calculated before proceeding to the next higher power level. A further use of the flux distribution was the calculation of geometrical distribution of fuel burn-up and its effect on reactivity. Ultimately, when all wires have been mounted, a complete picture of the space variation of flux in the MTR will be available under different operating conditions.

In a typical set of flux patterns obtained at three stages during the burnout, the maximum fission rate does not occur at the center line of the reactor, but is reached several inches below. The position of the maximum shifts upward as fuel is burned up and the rods are withdrawn. In the final exhausted reactor the pattern is fairly symmetrical about the center line in most fuel positions. The distribution for the fully poisoned reactor is fairly close to a cosine except at the ends of the plates.

Attainment of Full Power Rating

3-50 After the critical experiments had been completed, the control rods calibrated, the temperature coefficient measured, and a rough estimate of power obtained, preparations were made to take the reactor up to the design level.

The power was first raised to a value at which the servo system could be placed in operation. One servo ion chamber was inserted to the position of maximum sensitivity in its access hole. When this was done, it was found possible to place the reactor in automatic control at a power

in the neighborhood of 1000 watts. The entire system functioned smoothly, requiring only minor adjustments of the amplifier gain. There are two servo chambers in the reactor normally, although only one is used at any particular time. By adjusting the relative depths of these chambers, it was possible to have the reactor in automatic operation at any power level above 300 watts. As the reactor power was raised, the chambers were moved accordingly.

The power was increased to 250 kw, at which value the temperature rise in the water flowing at 1000 gpm was enough to permit a calorimetric determination of power within 10 per cent. The boron thermopiles in the permanent graphite, the N^{16} activity monitors in the exit water, and the neutron-level indicators, all of which are simply related to power, were calibrated against heat rejected to the cooling water.

Fuel-plate temperatures were calculated from the temperature differences across each element, and from the vertical flux distributions, measured with the cobalt wires, which gave a picture of the maximum-to-average fission rates. Since the integral under the fission-rate curve represents the total heat generated by the element, it was possible to normalize the flux curve to heat production by means of the temperature rise through the fuel element in question. Having determined at each step that the maximum plate temperatures were lower than the values which would give film boiling in the water, the reactor power was increased to the design power of 30 megawatts. There were frequent shutdown periods during the burn-up of this first fuel charge. These were due principally to magnet failures and the consequent dropping of the shim rods. Apparently, the leakage of water around the neoprene gaskets was sufficient to cause the insulation resistance to degenerate gradually until finally a point was reached at which the holding force of a magnet was not sufficient to support its shim rod with full water flow. This condition was corrected by eliminating the quick-disconnect feature and fastening the magnets permanently to the actuating rods.

Excess Reactivity in the 3×9 Loading

3-51 After the attainment of design power with the MTR, the reactor was run at full power until

the combined effects of fission-product poison and fuel depletion caused the reactor to die. This short operating time was much less than had been anticipated.

In order to make a precise measurement of the excess reactivity available in the 3×9 core, a fresh charge of fuel was loaded and enough poison was added to withdraw all shim rods fully with the reactor critical. The amount required to do this is a measure of the excess reactivity. Because the poison was added in discrete steps in this determination, a calibration of the shim rods was

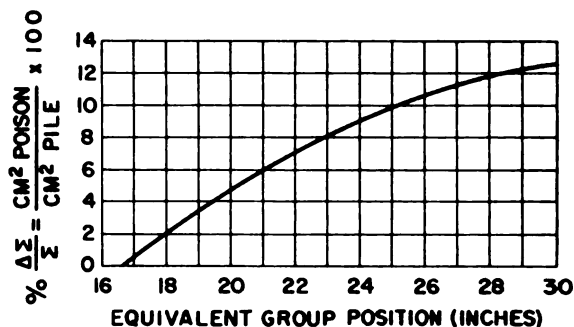


Fig. 3-45 Shim-rod calibration and excess k measurement for 3×9 lattice.

also obtained over their full travel. The shortage of suitable poisons made it necessary to use several materials. The final loading consisted of 31 silver, 6 gold, and 2 cobalt wires distributed as uniformly as possible in each of 23 fuel assemblies. Six poison points were obtained in moving the rods from 17 to 28 in. as a group. After several corrections had been made, the excess reactivity was estimated to be 12.8 per cent for this particular loading. Despite the fact that this excess reactivity is much less than the value of 18.9 per cent quoted in the "MTR Handbook," several percentage points were still unaccounted for. Several experiments were performed to explain the discrepancy.

Statistical Weight Experiments

3-52 An attempt was made to recalculate the reactivity loss due to burn-up. For a small heterogeneous reactor like the MTR, precise calculation of reactivities is very difficult, if not impossible, without resorting to semiempirical techniques. By dividing the reactor core into cells

and determining experimentally a weighting factor for each cell, one may then develop a convenient means for computing reactivity changes.

An approximate expression which emerges from perturbation theory gives Δk as

$$\frac{\Delta k}{k} = \sum_1^M \omega_i \Delta \Sigma_i$$

which is normalized so that

$$\Sigma \omega_i = \frac{M}{\Sigma_{pile}}$$

In this expression ω_i is the change in $\Delta k/k$ per unit of poison placed in the i th cell. If the poison is uniformly distributed throughout one fuel assembly, a statistical weight is obtained which is averaged in the vertical direction. Actually the flux distribution departs enough from a cosine in the vertical direction to make it desirable to have a vertical statistical weight averaged over some small distance, such as 1 in. Because of the large amount of time which would be consumed in doing this for every fuel assembly in the 3×9 lattice, it was decided to measure only the statistical weight averaged over an entire assembly. An experiment was performed in which a small amount of poison was moved from one lattice position to another for a number of selected fuel positions. In order that the calculations would apply to the reactor at the end of a burn-up, with all shim rods withdrawn to the upper limits, the statistical weight measurement was made on a fully poisoned reactor. Actually, this determination was made at the end of the poison calibration, in which enough silver had been added to retract all shim rods fully. Enough of the poison was removed from the 25 element (Fig. 3-38) so that the rods were slightly inserted with the reactor critical. By transposing the fuel element in 25 with the fuel assemblies in each of the other chosen lattice positions, the statistical weight of each position was determined relative to the 25 position. Figure 3-46 shows the map of the lattice. The numbers in brackets are interpolated from the measured values. For a cosine distribution the statistical weight should vary as the square of the flux. The values of the flux squared are divided by the flux squared of the 25 position and are shown for comparison in parentheses. The departure of this

quantity from the measured values of statistical weight is evident, particularly for the north row. This lack of agreement is ascribable to the non-uniform flux.

	1	2	3	4	5	6	7	8	9
1	(.40) .37	(.44) [.55]	(.04) [.73]	(.11) [.79]	(.05) .80	(.95) [.78]	(.88) [.70]	(.57) [.48]	(.38) [.27]
2	(.45) [.43]	(.62) [.60]	(.99) [.78]	(.92) [.92]	(.00) 1.00	(.78) [.92]	(.73) .75	(.38) [.52]	(.23) .30
3	(.47) .51	(.60) [.70]	(.04) .83	(.98) [.92]	(.02) .96	(.74) .91	(.57) .81	(.29) .57	(.19) .32

NUMBERS IN LATTICE POSITION SHOW MEASURED RELATIVE STATISTICAL WEIGHT FOR THAT POSITION. VALUES IN BRACKETS WERE ESTIMATED FROM MEASURED VALUES.

VALUES IN PARENTHESES WERE COMPUTED FROM TEMPERATURE RISE ACROSS ASSEMBLIES.

Fig. 3-46 Statistical weight map for 3×9 lattice.

Excess Reactivity and Statistical Weight of the 5×5 Core

3-53 As mentioned previously, the 5×5 array has a higher theoretical effective k . At this point, it was decided that measurement of the excess reactivity and the statistical weight for this array would be worth while. When the excess k was measured, it was found (Fig. 3-47) to be several

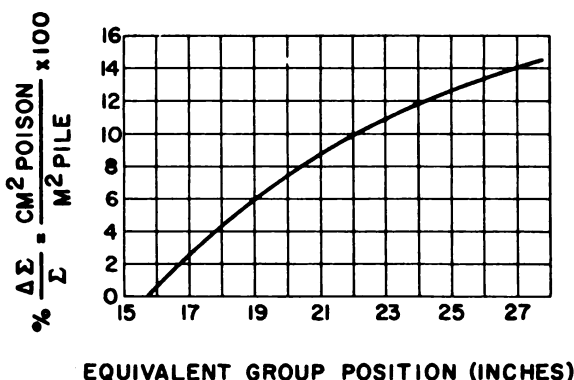


Fig. 3-47 Shim-rod calibration and excess k measurement for 5×5 lattice.

per cent higher than for the 3×9 loading, although there were 21 instead of 23 fuel elements in this array. The statistical weights were measured as before. Figure 3-48 gives the values obtained and estimated for the 5×5 array. Because of the higher inherent symmetry of this

	3	4	5	6	7
1	(.61)	(.67)	(.67)	(.61)	.50
2	(.79)	(.89)	(.90)	(.83)	(.68)
3	(.89)	(.99)	1.00	(.93)	(.78)
4	(.79)	(.89)	(.90)	(.83)	(.68)
5	.61	(.67)	.67	(.67)	.50

NUMBERS IN LATTICE POSITIONS SHOW MEASURED RELATIVE STATISTICAL WEIGHT FOR THAT POSITION.

VALUES IN PARENTHESES WERE ESTIMATED FROM MEASURED VALUES.

Fig. 3-48 Statistical weight map for 5×5 lattice.

configuration, it was felt that fewer points were needed.

Burn-up of the 5×5 Core

3-54 To see what portion of the extra reactivity in the 5×5 core was converted to lengthened operation, a new core in this configuration was run through a burn-up cycle. In Fig. 3-49 it can

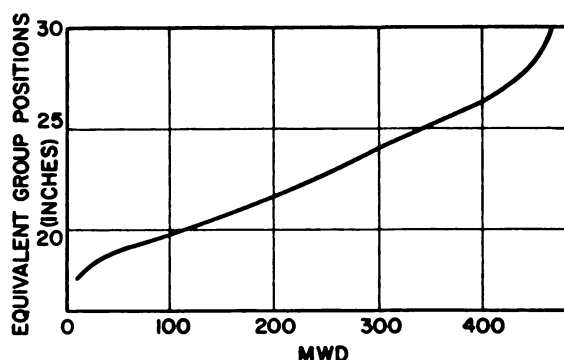


Fig. 3-49 Shim-rod position vs. megawatt-days of operation for 5×5 loading.

be seen that most of the additional reactivity was converted into operating time. A total of 467 megawatt-days was attained with this core. The operating history for the 5×5 core burn-up

reflects the improvement in operation since the first 3×9 run.

Reactivity Calculations

3-55 Using the new data for the reactor for such quantities as aluminum-water ratio, fuel content per assembly, distribution of power among the various lattice positions, and temperature coefficient, the excess reactivity in the MTR was recalculated. The results of these calculations, the experimental value, and the design values are given in Table 3-26.

Table 3-26 Distribution of Reactivity

	Design	Calculated	Experimental
$\text{Xe}^{135} + \text{Sm}^{149}$ (equilibrium at 30 Mw).....	4.9	4.9	
$\text{Xe}^{135} + \text{Sm}^{149}$ (override)....	4.7		
Fuel depletion and low-cross-section fission products...	3.5	7.0	
Temperature.....	0.8	0.3	
Others (experimental load)...	5.0		
Total.....	18.9%	12.2%	12.8%

Two principal effects which account for most of the difference between the measured reactivity and the design value are (1) the aluminum-water ratio, which was much higher than in the original critical experiment (0.72 compared to 0.65), and (2) the reactivity loss due solely to the spatially nonuniform burn-up of fuel.

Miscellaneous Experiments

3-56 The Effect of Xenon on the Operation of the MTR. In the design of the MTR, an allowance of 4.9 per cent reactivity was made to offset the effect of the equilibrium concentration of xenon and samarium at 30 megawatts. Because xenon 135 is a decay product of iodine 135, which is generated in an amount proportional to the fission rate, immediately after shutdown and the consequent cessation of burn-up of xenon the concentration of xenon begins to increase rapidly, reaching a maximum at about 11 hr after the power reduction. This maximum concentration corresponds to 43 per cent in $\Delta k/k$, which is far

greater than the excess reactivity built into the MTR. In order to provide some flexibility of operation, 4.7 per cent excess reactivity was allotted in design specifically for xenon override. This gives approximately 30 min in which to get back up to power after a transient shutdown.

Several xenon experiments were conducted at Oak Ridge on the LITR at a power of about 1 megawatt. To provide additional information on xenon at higher power and its effect on operating reactors, an experiment was performed on the MTR which resulted in a complete picture of the attainment of equilibrium xenon concentration, the build-up after power reduction, and the decay after the maximum concentration had been reached. An operating level of 5 megawatts was chosen for this experiment so that the peak xenon concentration after power reduction could be overridden with the reactor critical. Furthermore, the temperature rise at this power is quite small. This avoids difficult temperature corrections.

The experiment was done as follows: The reactor was loaded with a fresh 3×9 charge. Water flow was begun several hours before the experiment so that the entire reactor tank would be at a constant temperature. Power was raised to the 5-megawatt level and maintained there for about 48 hr, until the rod motion had settled down to the constant slope of samarium production plus fuel burn-up. When this condition was obtained, the reactor level was lowered to 50 kw and held there in order to follow the decay of xenon.

The $\Delta k/k$ is plotted as a function of time after start of the run. An equilibrium concentration equivalent to a reactivity loss of 3.6 per cent was reached about 40 hr after startup. The reactivity loss increased after shutdown to a maximum value of 8.6 per cent, which occurred 8.8 hr after the power was reduced. These values are not corrected for samarium poisoning and fuel burn-up, both of which contribute little under the conditions of this experiment.

3-57 Temperature and Pressure Measurements of Water from Fuel Assemblies. At various power levels, measurements were made, using the reactor instrumentation, to determine the pressure and temperature-rise profiles in the fuel assemblies. Temperature was measured by

iron-constantan thermocouples located just under each fuel assembly. Data for the 3×9 array are given in Table 3-27. The profiles for each row of fuel elements for the 28-megawatt case are given in Fig. 3-50. These temperature rises were used,

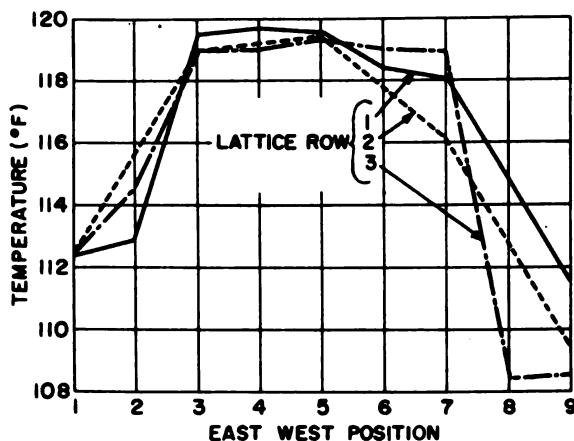


Fig. 3-50 Temperature profile at 28 megawatts for 3×9 loading.

together with maximum-to-average ratios in the horizontal direction obtained from these temperature data and the maximum-to-average ratios in the vertical direction obtained from the flux traverses, to calculate maximum plate temperatures in the fuel elements. In Table 3-28 the equivalent data taken in the 5×5 burn-up are given. Maximum plate-temperature calculations are not given in this table. In Fig. 3-51 a plot

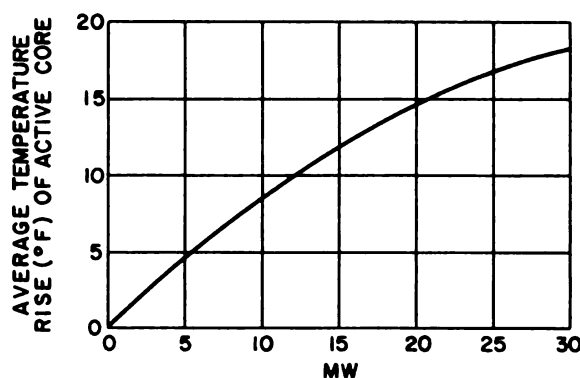


Fig. 3-51 Average temperature rise vs. power for 5×5 loading.

has been made of these and similar data to give an average temperature rise across the fuel elements as a function of power. Recalling that the

Table 3-27 Temperature Rises through Fuel Elements
(3 × 9 lattice)

Power, megawatts	20.1	20.2	22.2	24.9	26.1	28.0	29.8
Fuel-element position	Temperature rises, °F						
11	11.5	9.0	10.2	11.1	11.38	12.18	13.57
12	11	9.7	9.5	...	8.85	12.90	13.18
13	17	15.8	17.4	18.0	18.45	19.53	20.04
14	17.4	16.6	16.7	18.4	19.05	19.74	21.30
15	17.5	16.0	16.7	17.9	18.5	19.65	20.02
16	15.8	15.0	16.7	18.9	19.4	18.38	19.98
17	11.0	14.3	15.6	16.2	16.5	18.05	18.55
18	13	11.1	12.8	13.5	13.7	14.62	15.38
19	11.3	9.5	11.0	11.1	11.62	11.57	12.10
21	11.0	9.8	10.9	11.6	12.12	12.99	13.63
22							
23	14.5	15.1	16.9	18.5	18.9	18.98	19.96
24							
25	16.7	15.1	15.8	17.2	17.8	19.48	20.21
26							
27	15.2	12.6	13.2	15.0	8.9	16.12	17.45
28							
29	8.6	7.3	8.0	8.5	8.6	9.40	9.48
31	11.4	10.2	10.4	12.1	11.9	13.12	14.22
32	13.9	12.1	11.8	13.8	14.0	14.62	14.36
33	17.2	16.1	15.3	17.6	18.0	19.06	20.08
34	16.7	15.1	14.9	17.5	11.9	19.06	19.10
35	17	15.6	15.1	17.5	12.05	19.48	20.32
36	12.1	10.1	11.3	13.7	13.5	(19.06)	19.10
37	12.5	10.9	12.0	13.6	13.6	(19.06)	11.42
38	11.3	8.9	10.0	11.9	11.8	8.47	11.70
39	8.7	6.4	7.5	7.6	7.7	8.80	8.70
Avg ΔT , °F.....	13.6	12.3	13.0	14.6	13.8	15.8	16.26
Max ΔT , °F.....	17.5	16.6	17.4	18.9	19.4	19.75	21.3
Max/avg (horiz).....	1.29	1.35	1.34	1.30	1.41	1.25	1.31
Max/avg (vert).....	1.6	1.6	1.5	1.4	1.4	1.3	1.3
Water flow, gpm.....	17,900	18,950	18,900	18,900	19,000	19,150	19,200
Inlet temp, °F.....	100	90	90	92.6	90.2	90.7	97.2
Bulk water ΔT , °F.....	...	7.3	8.03	9.02	9.4	10.0	10.65
Max plate temp, °F (100°F inlet).....	201	200	199	202	205	200	208

Note: Figures in parentheses are estimated.

Table 3-28 Temperature Rises through Fuel Elements
(5 × 5 lattice)

Power, Mw	5	10	15	20	23	25	27	30
Fuel-element position	Temperature rises, °F							
13	4.2	7.9	10.9	13.4	14.5	15.1	15.5	16.0
14	3.9	7.4	10.6	13.2	14.5	15.3	15.9	16.7
15	4.5	8.7	12.5	15.6	16.9	17.6	18.1	18.5
16	4.3	8.5	12.2	15.3	16.8	17.5	18.1	18.6
17	4.5	8.6	12.1	14.9	16.3	17.1	17.7	18.4
23	4.3	8.2	11.6	14.4	15.9	16.7	17.3	18.1
25	4.3	8.3	11.9	14.7	16.0	16.6	17.0	17.3
27	4.0	7.8	11.4	14.4	15.9	16.6	17.2	17.8
33	4.5	9.0	12.9	16.0	17.4	18.2	18.7	19.1
34	4.4	8.7	12.7	16.1	17.8	18.8	19.5	20.3
35	(4.6)	(8.8)	(12.4)	(15.2)	(16.5)	(17.2)	(17.7)	(18.2)
36	2.9	5.9	8.7	11.3	12.8	13.7	14.6	15.8
37	4.3	8.2	11.6	14.5	15.9	16.7	17.3	18.1
43	(4.0)	(8.0)	(11.7)	(14.6)	(16.1)	(16.9)	(17.4)	17.9
45	4.9	9.2	12.8	15.7	17.1	17.8	18.4	19.1
47	4.0	7.8	11.1	13.9	13.3	16.1	16.8	17.6
53	3.6	7.1	10.4	13.3	14.8	15.5	16.2	16.8
54	(4.0)	(7.8)	(11.3)	(14.2)	(15.6)	(16.4)	(17.2)	(17.9)
55	4.4	8.5	12.1	15.1	16.5	17.4	18.2	19.0
56	4.4	8.3	11.5	14.1	15.3	15.9	16.5	17.1
57	4.1	7.8	11.1	13.8	15.1	15.9	16.6	17.3
Average	4.2	8.1	11.6	14.5	15.8	16.6	17.2	17.9
Max/avg	1.17	1.14	1.11	1.11	1.13	1.13	1.13	1.13

Note: Figures in parentheses are estimated.

power is defined on the basis of the average bulk-water temperature rise, one is led to believe that either relatively more heat is being generated in the beryllium at higher powers, the thermocouples are giving false signals possibly because of gamma-heat effects, or there is some effect, not now understood, at work. This matter is now being studied.

The impact and static pressures at the fuel-assembly exits were measured during the 5 × 5 run with the pitot tubes provided as part of the reactor instrumentation. These data, and the water flows calculated from them, are given in

Table 3-29. It will be seen that the flow distribution varies less than ±5 per cent over the entire lattice.

3-58 Gamma Heating. To provide information on the magnitude of the gamma-heat generation in the reactor at full power, a measurement was made in several of the vertical holes in the graphite pebble zone just outside the reactor tank and in certain positions inside the tank itself. These measurements will continue to be made for some time.

The instrument used consisted of a cylinder insulated on all but one end, with that end in contact with a lower temperature environment. As heat is generated in the sample, the temperature of the insulated end rises. The gamma-heat generation can be calculated from the temperature

losses were measured in the usual manner by measuring changes in critical positions of control rods. Table 3-31 summarizes these experiments. It may be seen that the withdrawal of all horizontal beam plugs is estimated to cost approximately 2.25 per cent in $\Delta k/k$. From this study it may be

Table 3-29 Static and Impact Pressures of Pitot Tubes
(5 × 5 lattice)

Fuel-element position	Impact pressure, psi	Static pressure, psi	ΔP , psi	Water flow, gpm
13	29.0	7.5	21.5	476
14	28.9	6.1	22.8	490
15	30.1	6.7	23.4	496
16	30.3	8.1	22.2	483
17	30.2	8.4	21.8	478
23	30.7	8.8	21.9	480
25	31.0	7.7	23.3	495
27	31.8	8.5	23.3	495
33	29.9	9.2	20.7	466
34	30.4	8.0	22.4	485
35	31.9	8.4	23.5	497
36	30.7	9.0	21.7	477
37	30.3	9.2	21.1	471
43	31.3	8.4	22.9	491
45	31.8	7.1	24.7	510
47	31.4	8.5	22.9	491
53	29.3	8.9	20.4	463
54	30.6	8.1	22.5	486
55	31.1	7.5	23.6	498
56	30.5	7.3	23.2	494
57	29.2	8.0	21.2	473
Reactor average.....			22.5	485

difference. Data on these measurements are presented in Table 3-30, together with comparable values from similar studies made in the LITR.

3-59 Reactivity Losses Due to Voids in Reflector (3 × 9 Array). A study was made at this point of the effect of voids on reactivity. The experimental beam-hole plugs were withdrawn various amounts from their normal fully inserted positions, leaving holes in the reflector. The k

Table 3-30 Gamma-heat Generation

Location	Depth	Power, megawatts	Heat generation, watts/gram of aluminum
A-41	20'6"	15	0.16
A-33	20'6"	30	1.1
VG-7	20'10"	30	0.090
VG-7	20'6"	30	0.041 *
VG-11 †	18'6"	30	0.017
VG-11 †	19'0"	30	0.028
VG-11 †	19'6"	30	0.041
VG-11 †	20'0"	30	0.055
VG-11 †	20'6"	30	0.076
VG-11 †	20'10"	30	0.083
VG-11 †	21'0"	30	0.084
VG-11 †	21'2"	30	0.083
VG-11 †	21'6"	30	0.076

* 5 × 5 lattice.

† Shim-rod positions 18.6 in.

concluded that even if cadmium were placed in each beam hole the loss would only be about 30 per cent more.

Table 3-31 Effect of Voids on Reactivity
(3 × 9 lattice)

Void	$\Delta k/k$ loss, %
HB-2 moved out 6 in.....	0.2
HB-2 fully withdrawn.....	0.5
All HB's moved out 6 in.....	0.5
All HB's fully withdrawn (estimated).....	2.25

Recommendations for Changes in Fuel Assemblies

At the time that it was determined how much excess reactivity was available in the MTR, it was decided that it would be relatively simple to provide additional k/k by increasing the uranium content of the alloy in the fuel plates. The metal-

lurgical studies which were made during the development of the MTR fuel assemblies had been carried to alloys of concentrations as high as 30 per cent by weight. By increasing the fuel content per assembly from 140 to 168 g, it was estimated that the excess reactivity would be 19 per cent. As a result an order was placed for one full loading using this higher concentration.

When the new fuel assemblies were received near the end of August, the irradiation program was interrupted to perform the tests necessary to determine if the expected gains in reactivity had been realized. These tests, which were similar to those conducted on the 140-g assemblies, were measurements of (1) the minimum critical mass, (2) excess reactivity, (3) flux traverses, and (4) the burn-up cycle, or operating time.

3-60 Critical Mass. The critical mass, determined substantially as before, was 1700 g. Because of the high neutron background present in the reflector at the time of this run, the multiplication was obscured for the first points and the precision of this measurement was impaired.

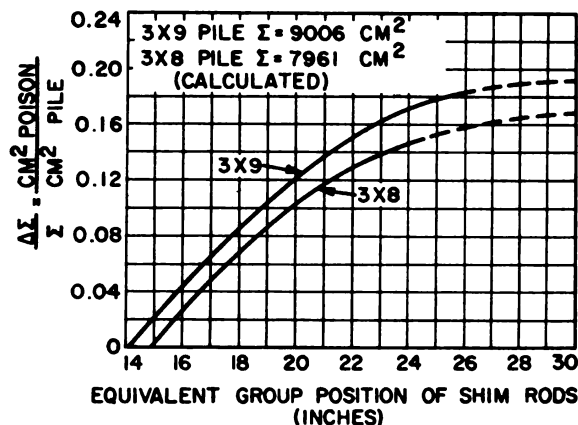


Fig. 3-52 Excess k and shim-rod calibration for 168-g fuel assemblies— 3×9 pile, $\Sigma = 9006 \text{ cm}^2$; 3×8 pile, $\Sigma = 7961 \text{ cm}^2$ (calculated).

3-61 Excess Reactivity. A complete silver-strip poisoning experiment was performed which yielded the information presented in Fig. 3-52. The excess reactivity was determined to be approximately 19 per cent for the 3×9 array and 16.5 per cent for a 3×8 array with beryllium pieces in positions 11, 21, and 31 (Fig. 3-38).

These values of excess reactivity indicated that the new assemblies would give a satisfactory operating cycle.

3-62 Flux Traverses. In Fig. 3-53 the vertical flux distribution in fuel element 25 is shown. A

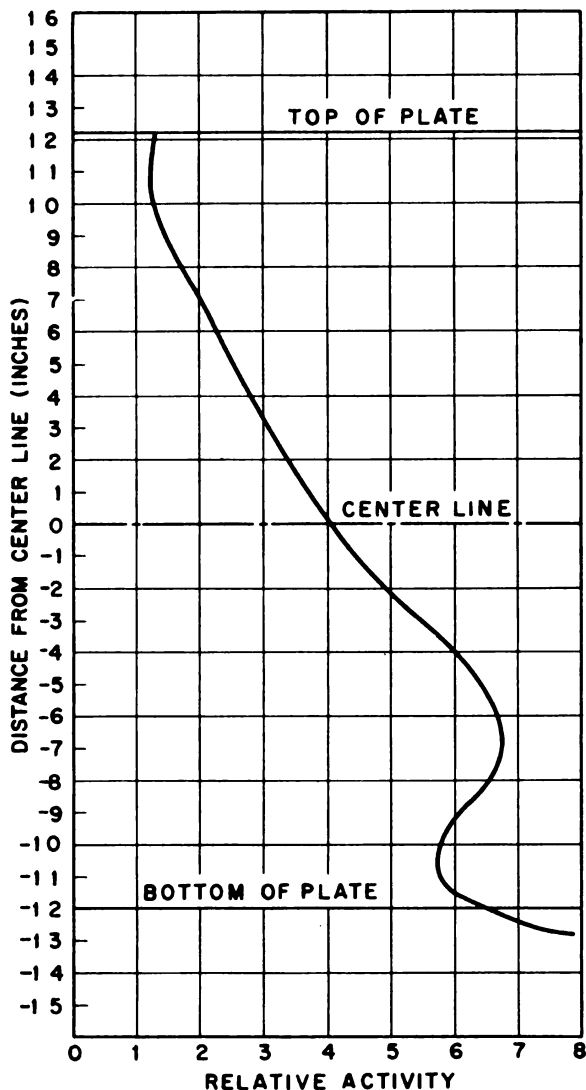


Fig. 3-53 Flux distribution in 168-g fuel assembly.

comparison with Fig. 3-43 shows that because of the lower positions of the shim rods the reactor is effectively smaller than before.

3-63 Burn-up. In Fig. 3-54 the plot of shim-rod group position versus megawatt-days of operation is shown. The total operating time was 625

megawatt-days, indicating that most of the extra reactivity went into additional operating time. With this run completed, the MTR was officially "in business" as a high-flux irradiation facility.

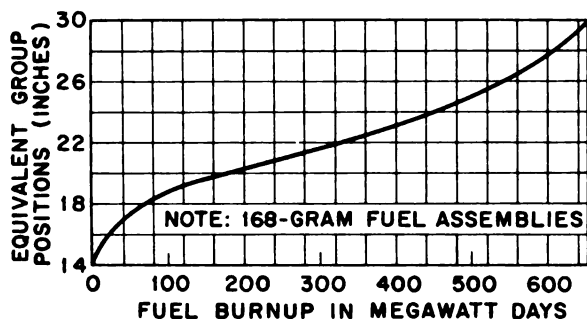


Fig. 3-54 Shim-rod position vs. megawatt-days of operation for 3×9 loading.

OBSERVATIONS AND CONCLUSIONS

3-64 The unique features of the MTR are thin-plate fuel assemblies, slab geometry, high specific power, and high excess reactivity. It is the first reactor having these features which has been operated above the 10^{14} thermal-neutron flux range. The success of the MTR is a milestone in atomic-energy work. The control system functions smoothly, and its speed of response is more than adequate to control any expected periods in this reactor. The experience obtained with the MTR proves that reactors having even higher excess reactivities are controllable.

An important result which came from the flux-mapping program was the demonstration of the large effect of the position of the shim rods on flux and power distribution. When the rods are one-half inserted, the MTR is essentially a half reactor. The flux pattern is quite nonuniform under such conditions, approaching a cosine distribution only when the rods are fully withdrawn. This nonuniform distribution increases the effective burn-up per megawatt-day appreciably. With this distribution, the power levels are less than they would be with a cosine distribution because the maximum-to-average plate temperatures are higher.

The one feature in the design of the MTR which has not proved satisfactory is the monitoring system which measures flow, temperature, and activ-

ity of the water under each fuel assembly. Readings of temperature have been erratic, and quite a few of the thermocouples have ceased functioning. The activity of the water as measured in an ionization chamber is masked by a high background due to accumulation of dirt from the water system in the chamber. The dynamic pressures, which are indicated by pitot tubes, are balanced in pairs to give annunciator signals for high or low differential readings. Because of the erratic behavior of these pitot tubes, it has been necessary to bypass this method of detection of water-flow anomalies.

Probably the most striking observation made during the startup program was the ease and relative smoothness with which the MTR ran through its paces in all phases of the program. This is attributable to the careful design which went into the MTR, to the meticulous checking carried on in the preneutron tests, to the extensive staff-developed skill at loading and unloading the reactor and at handling the controls.

Actually, many minor problems arose during the initial phases of operation. The most annoying difficulty was the behavior of the magnets which hold the shim rods to their driving mechanisms. These would often release the rods for no apparent reason and cause unscheduled scrams (emergency shutdowns). Considerable time was lost until water leaks into the magnets were eliminated. Spurious electrical fluctuations in the period channels caused unnecessary scrams until remedial steps were taken.

In the shielding some difficulty arose from excessive leaks around the key slots and annuli of the beam-hole plugs. These were easily eliminated by altering the key channels and by changing the outer shield biscuit from 8 in. of steel to the equivalent of 8 in. of lead. The effectiveness of the biological concrete shield speaks well for the "pre-pakt" method used in construction.

Other problems concerned the process-water system and were easily solved by minor changes in the design.

The postneutron tests were completed on Aug. 2, 1952. The Phillips Petroleum Company officially accepted the responsibility of operating the MTR on Aug. 11, 1952. Some experimental irradiations were started on Aug. 4, 1952. However, these were removed in order to permit a

series of tests in the last half of August to demonstrate that the excess reactivity predicted for the more concentrated fuel did exist. Full-scale insertion of experiments for other laboratories commenced on Sept. 4, 1952.

While a good deal of experience was obtained on the MTR during the acceptance tests, many interesting and important experiments remain to be done. Several of these have arisen as the result of operation of the reactor. A high background of neutrons exists in the MTR even with all fuel removed. This background has a number of short-lived components, but there also appears to be a net long-term increase. At the present time this seems to be chargeable to (γ, n) reactions in the beryllium reflector arising from activated impurities in the reactor. A consequence of this with regard to reactor design is that the magnitude and the diffuse nature of the background obscure the multiplication of the core unless chambers are located more favorably than in the MTR. While this causes no problem now, it might become one in the future.

Increasing the concentration of fuel has some-

what increased the reactivity in the regulating rod. A constant check is being made to see that the value of the regulating rod does not exceed desirable limits.

An unexpected radioactivity was discovered in the off-gases from the vacuum-spray evaporators in the process-water system. This has necessitated venting these off-gases to the main stack instead of allowing them to escape through the small stack on the Process-water Building. Preliminary investigations seem to indicate that this activity is due to gaseous fission products diffusing through the cladding of the fuel plates. There is no evidence of solid fission products in the process cooling water.

An evaluation of fission-product cross sections as a function of time should be studied. Other of the more obvious investigations which should be undertaken are long-term radiation damage, fuel-assembly development, induced radioactivities in structural materials, operation at higher power, and similar general problems in reactor technology where the high fluxes of the MTR permit study.

PART 4 OPERATING EXPERIENCE

This section concerns the operating experience with the reactor control and safety system and their components. The design philosophy and nature of the components has already been described in Part 1. This section will cover not only the troubles which were encountered and the revisions made to eliminate the difficulty, but will also indicate those ideas, components, or procedures which have proved particularly effective.

In order to visualize better how the various components are utilized in a high-flux reactor, a normal startup of the reactor will be described. Then the difficulties experienced with the various components will be described.

NORMAL STARTUP

3-65 A normal startup occurs after the regular shutdown work of changing fuel and experiments is completed. Present practice requires about 2 hr for this.

A startup check sheet (Table 3-32) is used to ensure proper setting and functioning of all facilities and safeties. Startup would not be dangerous without this check because control-rod withdrawal is automatically prohibited unless a certain number of safeties are operating, but time is saved in the long run if all parameters have the correct aspect and are known to be working before going to power.

After the top plug is bolted down and electrical connections are made, the shim drive rods are run in so that the magnets contact the armatures on top of the shim rods. During this time, full water flow and air flow are established. The foreman and supervisor are checking the logbooks, and safe work permits to see that equipment that has been worked on during the shutdown has been returned to its normal state. The reactor instrument group are going through their check list on switches and amplifiers in the reactor instrument room, and the reactor-console operator is checking the switches and instruments under his observation in the reactor control room.

The neutron source strength after 3 years of operation is several orders of magnitude higher

than during the original startup and is due to the γ - n reaction on the beryllium reflector. However, the power level is still only of the order of 1 watt, and the neutron level is below the range of all instrumentation except the count-rate meter. The operator moves the sensing element for this instrument (a fission chamber) and observes the change in reading to ensure that this channel is functioning normally. Startup is not permitted unless it is.

By this time the magnets have contacted the shim rods and the rods are raised an inch and dropped to check the relay scram circuits and the clutch and seat indicating system. Following this the rods are again raised a small amount to check the electronic scram circuits as far as this is possible without neutrons.

The reactor is now ready to "go critical," and the operator starts gang withdrawal of all shim rods on an intermittent basis—something less than 1 sec of withdrawal out of 5 sec imposed by an automatic timer. The operator knows approximately where the reactor will go critical from a calculation which takes into account the mixed loading and the effect of experiments. At first, the rod withdrawal has little effect on the reactor. Soon, the increased multiplication of the source can be seen. This effect eventually becomes great enough so that the increase with each intermittent withdrawal can be observed. By this time the level is within the log N and period range and the intermittent withdrawal generates false periods of increasingly short time constant. The reactor is almost critical, and smaller and smaller increments of $\Delta k/k$ are made to keep the period indication from getting shorter than about 20 sec. The reactor is now very responsive to rod movement, and on the next few increments of k the period meter fails to return to infinity and the reactor is above critical. The power is allowed to rise on a stable period of about 30 sec.

At 1 per cent of full power the automatic servo takes control and stops the rise in power. This halt would be made even in manual operation in order to observe the various experiments and reactor instrumentation. At this level all instru-

Table 3-32 Materials-testing Reactor Startup Check List

Date:

Time:

Engineer:

Logbook checked	Main water flow normal
Top plug in place	Bypass water flow normal
Bolted down	Air discharge pressure normal
Seal-water hoses connected	Instrument air pressure normal
Seal water on	Working reservoir level normal
Power cables connected	Sigma amplifiers normal
Regulating rods connected	Fuel element diff. flow normal
Process water and air flow requested	Activity normal
Beam holes' radiation doors closed	Temperature normal
Required facilities on	Argon flow normal
Neutron curtain in position	Nitrogen flow normal
Power off to winch	Desired period channel selected
Unloading mechanism clear	Scram reset
Subreactor-room doors closed	Shim-rod switches—one PREFERRED
Uninterrupted power checked	Others NORMAL
Power-distribution breakers closed	Desired shim rod SELECTED
Test-block covers on proper blocks	Magnet amplitude monitor on RESET
Instrument-room instruments on	Magnet currents normal
Instrument-cubicle instruments on	Regulating rods cocked
Monitor-room instruments on	Selected MOR at N_L
Control-board instruments on	Other MOR at N_F
Flux-level galvanometer on max sensitivity	Set point vernier in mid-range
Counting-rate galvanometer on	Check "manual scram"
Calibrate log N amplifiers	Check No. 1 level electronic scram
Level recorders read zero	No. 2 Level
Water-power recorder reads zero	No. 3 Level
N^{16} recorder reads zero	
Neutron-thermal recorder reads zero	
Period recorders read infinity	Clear annunciators
CRM off zero (fission chamber in)	
Count drops with fission chamber at CENTER	Safe work permits cleared
Return fission chamber to IN	

ments which have an appreciable reading at full power will now be giving an indication. Any which show an abnormal reading, either high or low, can be investigated or carefully watched during subsequent power increases. Roughly, 1 hr has elapsed since rod withdrawal was started.

Assuming that conditions warrant, the operator raises the demand on the servo, which brings the power up to the new level automatically on a 20-sec period. In a normal startup this is usually accomplished in approximately 5-megawatt steps. This gives time to outline experiments, to allow a redistribution of Xe in the lattice to avoid hot spots, and to do such things as add more shielding around the experimental facilities if required.

PERIOD INSTRUMENTATION

3-66 The system previously described for continuous monitoring of the reactor period has been found to have an exceptionally good long-term stability. The freedom from drift and component failures has been unusual. However, shortly after startup it appeared necessary to make some modifications to obtain short-term stability when the system is supplied from the small-capacity failure-free power source.

Trouble was experienced from the generation of false short periods concurrent with intermittent shim-rod withdrawal or starting of the amplidyne generator. Although the variations in voltage and frequency of the MG set were too small to be seen on the meters, as was the output from the ion-chamber power supplies, nevertheless the output varied sufficiently to charge the ion-chamber current. Bypassing the Sola transformer did not improve the situation, nor did the use of commercial power result in a stable period system although the effect of the shim-rod motor starting was reduced. The addition of 12 regulator tubes was sufficient to eliminate false periods resulting from transient loads on the MG power supply.

Although the addition of the regulator tubes was an improvement, the period meters were still quite nervous, resulting in too frequent period scrams because at that time the reserve holding power of the magnets was so low that a small rise in sigma bus voltage was likely to drop a rod. The wandering of the period meters was traced to variations in filament voltage on the log N diodes

due to drift in frequency of the MG set. Commercial power in this area is quite unstable; and while this in itself caused little trouble, the high-speed correction put in by the frequency controller on the MG unit would change the diode current at a rate sufficient to generate a fairly short period. The cure for this influence on period-system stability was to slow down the action of the mechanical part of the frequency controller.

The two charges mentioned resulted in fairly steady period recorders; even so, about once each day a period trouble indication and fairly frequent period scrams of such short duration would occur that none of the recorders could record even the slightest indication of an abnormal condition. This happened too frequently at 0800 hours to be considered a random fluctuation somewhere in the system. Although it was not possible to locate the source of the disturbance or establish the means whereby it was fed into the period system, it seems probable that it was a switching transient on the commercial power system which entered the period system, possibly by induction. At any rate, the addition of an integrating circuit to the period amplifier, at power, which gives at least a 20-to-1 protection against pulse-type interference has resulted in no period scrams since making the installation. The integrator is inserted into the circuit when the power level reaches $3N_L$; conversely, it is removed from the circuit as power drops below $3N_L$. At power levels above $3N_L$ when the integrator is in the circuit, a 1-sec period can cause an electronic scram only if its duration is greater than 1 sec.

CONTROL-ROD SYSTEM

3-67 The magnets supporting the shim-safety and control rods were a source of some difficulty. The original design was somewhat complicated by the limited space available. However, the initial tests showed that they had adequate strength to hold the rods with a 40-lb differential water pressure across the lattice and still have a release time less than 30 msec under stagnant water conditions. Under actual operating conditions the turbulent water conditions generated transient forces which exceeded those anticipated and which led to frequent dropping of a shim rod.

The immediate solution to the problem was to

increase the magnet current from 100 to 150 ma for each magnet. This increased the release time somewhat, but since the reserve holding power was still less than the design figure under full flow conditions, the actual release time was well within the 30 msec required to provide adequate safety.

Another consequence of the higher than anticipated forces was the leakage of water into the magnets. This was due to a weakness in the quick-disconnect mechanism which could not maintain adequate pressure on the flat gasket seal at the top of the magnet can. The original idea of the quick-disconnect mechanism was to keep down the exposure for the mechanic who would have to change a highly radioactive magnet after it had been exposed to full power flux for some time. Experience has shown that the magnets do not become excessively radioactive when they are normally positioned above the lattice; so the quick-disconnect mechanism was abandoned in favor of a bolted connection.

This change considerably reduced the problem of water entry into the magnet but did not completely eliminate it. A slight amount of water should not be objectionable since process water is maintained at about 1.5 μ mhos conductivity and the magnets are filled with an oil having a density greater than 1. However, under long use, with a slight amount of water there seemed to be an electrolytic or a galvanic action involving the silver plating on the lead-wire connectors in the connecting plug at the top of the magnet case. The loss of the silver eventually created a short along the top of the plug insulator and dropped the rod. The solution to this problem was to eliminate the connector plug entirely and make a direct connection to the magnet coil leads.

The one remaining problem which prevented completely reliable magnet operation was the magnet amplifiers themselves. While the components in these amplifiers operate considerably below rating, increasing the current to 150 ma does operate some of the components closer to their rating and increases the drift rate, adjustments required, and component failures. Under these conditions there have been times when one amplifier failed to take over the full load on failure of its mate and a dropped rod resulted. In the MTR, there is a fair portion of the operating cycle

during which power cannot be regained after dropping a rod because of the rapid growth of Xe poison and it becomes economic to secure a magnet-amplifier system which functions independent of component failures. The solution in this case was to add a third magnet amplifier to each magnet channel. Not only does this reduce the burden on each amplifier, but the chance of simultaneous failures in all three amplifiers is extremely remote.

Some operating difficulties have been experienced with the so-called clutch system. This system is the mechanism used to sense whether the shim drive rod is connected to the shim rod or whether the rod has dropped off. It depends for its function on the presence of a continuous electrical circuit to the insulated magnet case and through the shim rod and its bearings to the grounded lattice and tank structure. It is a low-voltage circuit for safety and is designed to "fail safe"; i.e., it will never tell the operator that a rod has been dropped unless it actually has dropped.

The difficulty has been that over a period of time the resistance between shim rod and ground has been increasing. This is due mainly to surface corrosion on the shim rod itself because the shim rods are used for extended periods and high burn-up. The result is that currents are reduced below the point of reliable relay operation, which results in dropping a rod on occasions, particularly during start or restart operation when the rods are moving. Some relief has been obtained by replacing bearings and using more sensitive relays. The point of diminishing returns in the latter has probably been reached because the sensitive relays are in themselves somewhat unreliable. The problem has been alleviated to some extent by using the seat relays as an electrical seal on the clutch system, but this complicates the system, which is objectionable. As a general rule it has been found that the simpler the system, the better the chance of recovery in case of system trouble.

The better solution to the problem is probably the use of magnetic amplifiers as the primary sensing element in the clutch-relay system. One complication that will then arise is the deterioration of insulation between the magnet case and the grounded drive rod. This insulation level is

now down to about 2000 ohms and places an upper limit on the sensitivity of the clutch-relay system. However, this deterioration does not appear to be a consequence of radiation damage and should present no real problem.

One anticipated problem that has not materialized is corrosion of the soft-iron armatures on top of the shim rods. These armatures are high-purity iron to obtain a high flux density before saturation and are nickel plated. A thin plating seems to stand up in this service quite well, the major damage occurring from mechanical handling. The armatures are reused and are not too radioactive to be replated in commercial shops.

The regulating rods have had two revisions since the original installation. These rods have no fuel and only limited $\Delta k/k$ change and are driven by the automatic servo system. They are stainless clad cadmium, constrained in the horizontal direction by stainless-steel sleeve bearings. As the flow rate of cooling water through the reactor has gradually been increased to provide cooling for the increased number of experiments in the Be reflector, side thrusts have developed that are sufficient to gall the rods. This problem was solved by putting nylon inserts in the sleeve bearings. Another problem with these rods was burn-out of the cadmium. The cadmium sleeve was increased from 20- to 40-mil wall thickness to extend the useful life of the rods from about 6 months to approximately a year.

SHUTDOWN PROCEDURE

3-68 The construction of the MTR requires that the reactor be shut down and water flow reduced to less than 1000 gpm when it is necessary to change fuel or experiments. The water pressure at the top plug during operation is about 70 psi. The normal procedure at shutdown is to scram the reactor by dropping the shim rods, then reduce the water flow and unbolt the top plug. In practice it is desirable to take advantage of each shutdown to proof-check the various reactor safety circuits as a method of scheduled shutting down. Instead of merely scrambling the reactor manually, a detailed safety-circuit check schedule has been developed to test each setback, reverse and scram, while serving as a double check on the routine electronic checking that is done during

shutdown. A safety-circuit check schedule is shown in Table 3-33.

After the reactor is shut down, the top plug containing the shim-rod drives and magnets, regulating-rod drives and associated drive motors is lifted from the reactor top by the overhead building crane and placed in a so-called dry dock on the first floor. The top-plug flange rests on the first floor, and the shim- and regulating-rod drives extend through a hole in the floor into the basement. This system allows for easy access to the rod drive motors and also to the shim-rod magnets during shutdown. Despite the improvements made in the motors during the last few years, it is still necessary to change magnets on the average of once every 6 weeks.

During shutdowns the overhead process-water storage tank is shut off from the rest of the cooling system by closing the block valve in the main 30-in. line to the reactor. Water circulation is continued by a small pump at the seal tank in a closed loop between the seal tank and the reactor bypassing the flash evaporators, sump tank, and sump-tank pumps. The maximum flow in this system is 1000 gpm, but after the reactor has been shut down for several hours, it is not necessary to circulate more than 200 to 300 gpm to keep the afterheat of the fuel elements from warming the reactor water appreciably. No cooling system is required in the seal-tank loop because heat loss from the tank and lines is more than adequate to prevent a measurable temperature rise. During some phases of shutdown work it is necessary to stop all water flow through the reactor. After 3 or 4 hr the tank water becomes warm enough thermally to cause some difficulty in seeing through the water. Convection currents of warm water coming up from the fuel elements cause density gradients and make visibility poor. On several occasions the water has become warm enough to cause a small amount of steaming, but it is doubtful that the water temperature rose higher than 130 or 140°F because of the heat loss from the surface and through the tank walls. In general the afterheat problem at the MTR has not been of any consequence.

Before the reactor top plug is removed, it is usually desirable to flush the contaminated process water out of the reactor and replace it with fresh demineralized water. This reduces the

Table 3-33 Safety-circuit Check Schedule to Be Performed as the Initiation of Routine Shutdown

Description	Date	Engineer
<p>In general, the shutdown will be accomplished in two steps:</p> <ol style="list-style-type: none"> 1. A reverse, fast, or slow setback, or combination of these 2. A scram from each circuit that causes a slow scram (to be initiated as soon as it is obvious that the first safety has functioned properly) <p>General check: Check that appropriate annunciators alarms and <i>both</i> slow scram lights come on</p> <ol style="list-style-type: none"> 1. Manual switching <ul style="list-style-type: none"> Reverse: Move reverse-bypass switch to REVERSE All shim rods insert Scram: Release reverse-bypass switch and move raise cock scram off NORMAL All shim rods drop 2. Boron thermopile <ul style="list-style-type: none"> Setback: Turn off boron thermopile recorder and move pointer just above $1.1N_F$ Slow setback Move pointer just above $1.2N_F$ Fast setback Scram: Move pointer above $1.5N_F$ All shim rods drop 3. Water power <ul style="list-style-type: none"> Setback: Turn off water-power recorder and move pointer above $1.1N_F$ Slow setback Move pointer above $1.2N_F$ Fast setback Scram: Move pointer above $1.5N_F$ All shim rods drop 4. N^{16} activity <ul style="list-style-type: none"> Setback: Turn off N^{16} recorder and move pointer just above $1.1N_F$ Slow setback Move pointer above $1.2N_F$ Fast setback Scram: Move pointer above $1.5N_F$ All shim rods drop 5. No. 1 level recorder and power failure <ul style="list-style-type: none"> Slow setback: Turn off No. 1 neutron level recorder and move pointer just above $1.1N_F$ Selected MOR starts slow setback Reverse: Move pointer above $1.2N_F$ All shim rods insert Scram: Turn off breaker 9 in motor-power distribution cabinet All rods drop Relay bus annunciator alarms 6. No. 2 level recorder and main flow valve <ul style="list-style-type: none"> Slow setback: Turn off No. 2 neutron level recorder and move pointer just above $1.1N_F$ Selected MOR starts slow setback Reverse: Move pointer above $1.2N_F$ All shim rods insert Scram: Allow reverse to carry power level below $3N_L$. Pull test block on main flow recorder. Make certain bypass flow is normal. Close motor-operated butterfly valve in main flow line All shim rods drop Sump pump 30-sec annunciator alarms 		

N_F = normal full power; N_L = 1% N_F .

Table 3-33 Safety-circuit Check Schedule to Be Performed as the Initiation of Routine Shutdown (Continued)

Description	Date	Engineer
<p>7. No. 3 level recorder and experiment Slow setback: Turn off No. 3 neutron level recorder and move pointer just above $1.1N_F$ Selected MOR starts slow setback Reverse: Move pointer above $1.2N_F$ All shim rods insert Scram: Initiate scram from experiment which is connected into scram circuit</p> <p>8. Low air flow and main flow valve position Slow setback: Cut off one fan in the Fan House Selected MOR starts slow setback Scram: Allow power level to decrease below $3N_L$. Pull test block on main flow recorder and then close downstream block valve All rods drop Sump pump 30-sec annunciator alarms</p> <p>9. Main water flow Slow setback: Close main flow-control valve so that flow is less than 90 per cent but greater than 80 per cent normal flow. (It may be necessary to reduce power slightly to prevent a setback from 105 per cent ΔT.) Selected MOR starts slow setback Scram: Before the power level reduces to $3N_L$, close main flow valve so that flow is less than 80 per cent of normal flow. (It will be necessary to use the valve on which there is no scram control for this. <i>Be sure to reopen and lock valve.</i>) All rods drop</p> <p>10. ΔT and bypass flow Slow setback: Turn off ΔT recorder and move pointer above 105 per cent normal Selected MOR starts slow setback Scram: Allow setback to carry power below $3N_L$. Pull test block 11 (sump pump 30 sec). Reduce main flow below 80 per cent normal. Close valve on bypass flow Scram at approximately 750 gpm</p> <p>11. Fast scram check Setback: Check an experiment having a reverse or fast or slow setback Scram: Have instrumentman put a volt box on sigma bus and record voltage reading as individual rods drop</p> <p>12. Loss of power to sump pumps Setback: Same as for step 11 Scram: Cut both feeders to Process-water Building (at the substation) Scram after approximately 20-sec delay. Check that butterfly valve closes. Check annunciator</p>		

spread of contamination when handling tools are removed from the tank. The reactor flush is accomplished in the seal-tank reactor loop. This loop has a capacity of approximately 40,000 gal. The water that is flushed out of the system is discharged to the retention basin for eventual discharge to the desert.

With the great number of reactor safety circuits in existence and the increasing number of large experiments which require power reduction when an experiment is not operating correctly, it

is not unusual for accidental shutdowns to occur. Table 3-34 summarizes the number of reactor shutdowns which have occurred during the first 3 years of operation.

After the reactor has been shut down by dropping the shim rods, the flush is completed and the top plug removed, it is possible to remove the upper assembly grid from the reactor and move it to the canal. This grid is somewhat radioactive from contamination and induced radioactivity, but does not usually measure more than 5 r/hr

at 10 ft in air. This level of radiation is not a serious hazard since the entire operation after the initial connection of the handling tool can be done from the overhead crane—a distance of over 40 ft.

Table 3-34

	1952	1953	1954
Number of scrams from reactor operation:			
Due to personnel failures:			
Inattention.....	0	0	0
Accidental activation.....	0	1	2
Due to equipment failure:			
Power supply.....	4	2	1
Instrument failure.....	5	5	5
Mechanical failure.....	13	4	9
Total.....	22	12	17
Number of scrams from experiment operation:			
Due to personnel failures:			
Inattention.....	0	0	0
Accidental activation.....	0	0	1
Due to equipment failure:			
Instrument failure.....	0	3	16
Mechanical failure.....	0	3	9
Total.....	0	6	26
Total.....	22	18	43

The working platform is then placed over the reactor, and the fuel and experiments in the reflector are easily moved into the unloader and transferred to the canal for storage. Since the operators are working straight above the reactor core and reflector and are looking through a column of water about 18 ft thick, the handling tools do not need to be complicated. Most of the tools in use have been developed as they were required. None of these are so unique as to deserve detailed description.

The water column above the core is quite adequate to protect personnel from excessive radiation during shutdown work, at least from radiation coming from the fuel or experiments. Occasionally after experiencing a fission break which allows appreciable amounts of fission products or U to enter the process-water system, the radia-

tion level at the reactor top is above tolerance. This is due to the adsorption of fission products on the walls of the tank above the shutdown water level. The problem of fission products in the process-water system is covered more fully in another section.

Refueling of the reactor is a relatively simple job and requires only about 3 to 5 hr to complete. Most of the reactor shutdowns which are now scheduled last 70 to 100 hr. This extension is entirely due to the changing of experiments both in the reactor tank and in the beam holes.

After the reactor is brought up to power and fission-product absorptions have come to equilibrium, the reactivity loss in the core is mainly due to fuel burn-up: To compensate for this, it is necessary to withdraw the shim rods periodically. Withdrawal in steps is permissible because the regulating rod will act so as to maintain a constant flux in its chamber and therefore will maintain nearly constant power in the reactor.

Near the end of the run the shim rods are completely withdrawn, leaving the reactor in control by the regulating rod alone. Because of its servo system, the regulating rod will withdraw continuously to restore any loss in reactivity resulting from burn-up.

Occasionally it is desirable near the end of a run to reduce the inlet water temperature as much as possible to add reactivity to the reactor, thus increasing the lifetime of the fuel charge by a few hours. By observing the induced change in regulating-rod position, this change in water temperature afforded an excellent opportunity to measure the temperature coefficient of the reactivity of the reactor at full-power (30-megawatt) operation.

STORAGE OF RADIOACTIVE PARTS

The various types of assemblies discharged from the reactor are stored in racks bolted to the floor at the end of the canal. The racks are designed for easy insertion and removal by providing unrestricted accessibility from the side. The short assemblies are stored vertically or inclined at a slight angle to the vertical, while the long rods are stored parallel to the canal floor or at an incline to the horizontal. The canal purge is utilized to cool the assemblies.

3-69 Beryllium Assemblies. The storage of the beryllium-reflector assemblies presents no problem except that they must remain adequately shielded. An aluminum rack provides for the storage of 36 of the assemblies.

3-70 Shim-safety Rods. The length of the shim-safety rod prohibits vertical handling unless shielding is provided in addition to that given by the canal water. To ensure thermal siphoning and adequate shielding, the rods are stored at a 15° angle to the horizontal. They remain in this position until sufficient decay time has elapsed to permit the rods to be sawed into shorter lengths for transfer to decay storage or to chemical processing. The rods are stored with the bottom section at the lowest elevation. The canal purge dissipates the heat generated by the assemblies.

The racks for the storage of shim-safety rods are limited to groups of 16, four in a row and four tiers high. One rack has been installed, while the future installation of racks in the other side of the canal is provided for by expansion anchors for the racks in the floor of the canal.

The rack consists of three sections, each of which is bolted to the canal floor and so located that the rod rests in all three. The end sections are U shaped to contain the rod and to permit easy insertion and withdrawal. The lower end section is closed at the bottom to stop the rod. The middle section supports the uranium section of the shim-safety rod. It is constructed in an angular shape and is lined with cadmium sheet $\frac{1}{32}$ in. thick to absorb neutrons.

3-71 Regulating Rods. Four regulating rods are stored parallel to the floor in U-shaped openings constructed at the bottom of the shim-safety storage rack. These openings are placed on the inside and outside of two adjacent shim-safety rods; this permits easy insertion and withdrawal of the rods.

3-72 Fuel Assemblies. Two types of racks are provided for storage of more than 100 fuel assemblies. One type rack is an A frame fabricated from stainless steel. The sides and back of the pocket for the assembly are lined with thin cadmium sheet. The front is open toward the side wall or center of the canal for easy accessibility.

Four holes in the bottom of the pocket permit water to flow into the assembly to keep it cool by thermal siphoning.

The other type of storage rack is designed in the form of a grid to provide an irradiation facility using the spent fuel elements as the radiation source. Details of this grid are shown in Appendix B at the end of the chapter. This type of storage configuration provides for three irradiation facilities, each $3\frac{1}{2}$ by $12\frac{3}{4}$ by 37 in. high, and one facility $3\frac{1}{2}$ by $7\frac{3}{4}$ by 37 in. high. A description of the use of this grid is given in Secs. 3-74 to 3-78.

GENERAL PROBLEMS ENCOUNTERED AFTER A LONG OPERATING PERIOD

3-73 During the operation of the MTR, several minor fission breaks occurred which released some uranium and fission products into the process water. The short-time effects of these fission breaks were not hazardous; i.e., they resulted in no high radiation levels; however, the long-term effects of these fission breaks affected operation for some time by raising the level of activity in the entire process-water system and reactor tank.

Considerable experience has been gained during the operation of the MTR in the reliability of fission-break monitors. It is perhaps worth listing the points at which the system is monitored, before any trouble occurs, and the drawbacks of these monitors.

1. Individual monitors on each fuel element. This system samples water from each of the fuel elements and after a delay time sufficient to decay the N^{16} activity sends it through an ion chamber. Each fuel-element activity is recorded separately.

2. High-pressure ion chamber on the main process-water stream about 1 min after the water leaves the reactor.

3. High-pressure ion chamber on the gases vented from the ejector unit.

4. Weekly samples of process water, evaporated and counted.

5. Several scintillation-type monitors on the effluent from various experiments.

6. High-pressure ion chambers on the gaseous and particulate activity leaving the stack.

When fission products were released into the MTR system, it soon became obvious that none of the above monitors was satisfactory, with the possible exception of some of the monitors on the experiments. Because the fission breaks did not occur on the monitored experiments, none of the monitors was sufficient, as their only value was to tell us that the activity was abnormally high.

Fission breaks at various times resulted in the following activity problems:

1. High gaseous activity discharged from the MTR stack resulting from fission-product gases removed from the process water by the steam ejector unit.

2. High gaseous activity in the Process-water Building operating area. This was intermittent and depended upon wind conditions.

3. Greatly increased activity on the pipe walls of the process-water system and in the MTR tank. Activity as high as 3 r/hr was obtained at contact on the outside of tanks and pipes in the MTR process-water system whereas it had been about 50 to 100 mr/hr before fission breaks occurred. Activity as high as 5 to 7 r/hr was obtained inside the MTR tank where it was necessary to work in order to change a number of the experiments. This activity of course died off to a great extent after the source of activity was removed from the system and normal operation was resumed; however, on a short-term basis it did create an operating problem by limiting working time in the MTR tank and during maintenance of the process-water system. Normal operation of the process-water system was not a problem because all the system is shielded.

4. Necessity to de-gas for 3 to 4 hr before removing the top plug after reactor shutdown, because of the gaseous activity in the process water. This procedure was necessary to avoid radioactive contamination of the air in the Reactor Building.

5. High gaseous activity in the Reactor Building. This resulted occasionally if wind conditions were unfavorable.

6. Plating of uranium on the reactor parts. This was the most serious effect of fission breaks

in the MTR because it resulted in long-term activity problems when the continued operation of the reactor caused fissioning of the U^{235} plated on the reactor parts. It was observed that the activity plated more on the aluminum parts than on the stainless steel, as noted below.

The program of investigation of process-water activity has been extended to include an analysis of material deposited on the metal surfaces of the reactor water system. Gamma analysis of the iodine fraction of the process water indicated constant age distribution (relative isotopic abundance) which is characteristic of fission-product recoils. This suggested the exploration of the possibility that the fission products being introduced into the process-water system might be largely due to recoils from uranium deposited on the metal surfaces in the lattice. Samples of beryllium, aluminum, and stainless steel were obtained from positions both in and out of the active lattice. The outer surface was removed from these samples and irradiated in the reactor. Radioanalysis for fission-product iodine was utilized to determine the uranium deposited on the surface of each sample. Table 3-35 summarizes the results to date.

Table 3-35 Uranium Deposition on Metal Surfaces in the Reactor

<i>Sample description</i>	<i>Uranium, g/cm²</i>
Stainless coupon from active lattice.....	3×10^{-8}
Aluminum coupon from active lattice.....	6×10^{-7}
Beryllium coupon from active lattice.....	1.5×10^{-7}
Stainless tubing from reactor tank.....	2.0×10^{-10}
Aluminum tubing from reactor tank.....	1.7×10^{-9}

Although these results are preliminary in nature, it is apparent that the deposition rate on aluminum is considerably above that of stainless steel and beryllium. Also, there is an indication that the deposition rate is higher in the active lattice. This deposition is responsible for the constant level of activity now being found in the water.

PART 5 GAMMA FACILITY

3-74 Radiation Sources. In the process of U^{235} fission in the reactor, probably several hundred radioactive fission-product species are produced. In addition to the prompt beta and gamma radiations which follow a $U^{235}(n,f)$ reaction, the fission products continue to give off beta

cance because of their inability to penetrate any appreciable thickness of water. The spent fuel element is therefore considered to be a source of relatively pure high-energy gamma rays.

Probably a hundred radioactive fission-product species have been reported to emit gamma rays in their decay. However, because of low fission yield, low energy levels, or short half-lives, the nuclides of interest are reduced to perhaps a score. Since the decay schemes of most of these are known, it should be possible to determine the energy spectrum of the emitted gamma rays for any time after shutdown. Reported values of the average energy from this type of radiation vary from 0.8 to 1.6 Mev, with 1.0 Mev being a likely value. The average energy and the intensity drop off with time. This does not impose any serious obstacle in the use of the spent fuel elements as a radiation source because the average energy remains relatively constant for quite long periods and the radiation intensity can be readily measured and converted with reasonable accuracy to dosage values.

Curve II in Fig. 3-55 represents the decay of a typical individual fuel element measured under water at a fixed distance from the center line of the element. Curve I presents data taken from one of the grid facilities, giving a composite effect of several elements which most likely have different reactor histories as to position, amount of U^{235} burn-up, and possibly even to initial quantity of fuel and cooling time. Since the grid facilities are rectangular, the contributions from elements vary. It is noted, however, that the slope of the grid radiation compares reasonably well with the single element. The saw-tooth nature of the grid facility is caused by replacing older elements with fresher elements removed each cycle from the reactor. The curves shown in Fig. 3-55 should be taken as representative of typical types of decays; the absolute values are only indicative of the order of magnitude to be expected.

3-75 Gamma-radiation Measuring Devices. It is apparent that, for any quantitative assess-

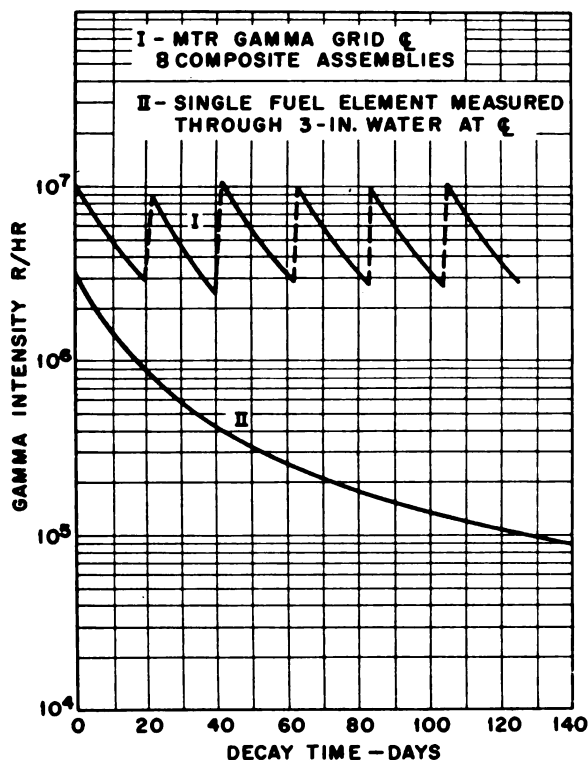


Fig. 3-55 Typical decay curves—intensity (r/hr) vs. time (days).

and gamma rays for some time after the reaction. The spent fuel element, after removal from the reactor, therefore becomes a logical radiation source. In addition to the beta and gamma radiation, a neutron flux of the order of 10^4 neutrons cm^2/sec is found around a discharged fuel element. This flux, attributed primarily to the (γ,n) reaction of the 2.5-Mev gamma rays from La^{140} with the deuterium in the water, is considered small in comparison to the other radiation. The negative beta particles emitted from the discharged fuel element are also considered to have little signifi-

ment of radiation dosage, intensities must be known for individual flux components. Sensitive, reliable instruments are therefore an important adjunct to any irradiation work. The following three methods have been used satisfactorily for gamma-intensity measurement. There are no instruments, except the calorimeter and perhaps the ion chamber, regularly available at the MTR for use in conjunction with gamma experiments. All instrumentation required in any gamma experiment, except the above, must be provided by the experimenter.

Chemical dosimeter. One of the older methods of gamma-radiation intensity measurement, but one which has not been widely adopted, consists of exposing a solution of ceric sulfate for a fixed time, then determining the amount of cerous ion formed. The change in concentration of the ceric ion can be determined by volumetric analysis or by a colorimetric procedure. The yield is independent of the concentration from 3.2×10^{-4} to about 10^{-5} molar and independent of gamma energy from about 100 kev to 2 Mev. This method has given reliable results up to 3×10^7 r/hr and is still used as a standard at the MTR. However, it has the disadvantage of being slow.

Rod-type calorimeter. Figure 3-56 shows a schematic drawing of this instrument. It is primarily a center cylinder of stainless steel encased in an outer shell of aluminum. Thermocouples are silver soldered into the ends of the cylinder. The cylinder is thermally insulated with air on all surfaces except one end. In this arrangement, heat induced in the cylinder by gamma activity can be dissipated only toward the lower end. This creates a temperature differential between the two thermocouples, the magnitude of which is proportional to the gamma activity. Calibration of this instrument against the chemical dosimeter permits conversion to roentgen per hour units. This instrument is faster and more versatile than the chemical dosimeter, but still not so rapid as the ion chamber.

Ion chamber. Figure 3-57 shows a sketch of the air-aluminum ion chamber constructed for use in the MTR gamma facility. The active volume is about 11 cm³, with 900 volts applied between probe and wall. This voltage was found to be more than adequate to reach saturation current. The instrument has given good correlation with

the ceric dosimeter and has the advantage of being rapid.

Silver-activated phosphate glass dosimeters have been used for measuring gamma dosages at

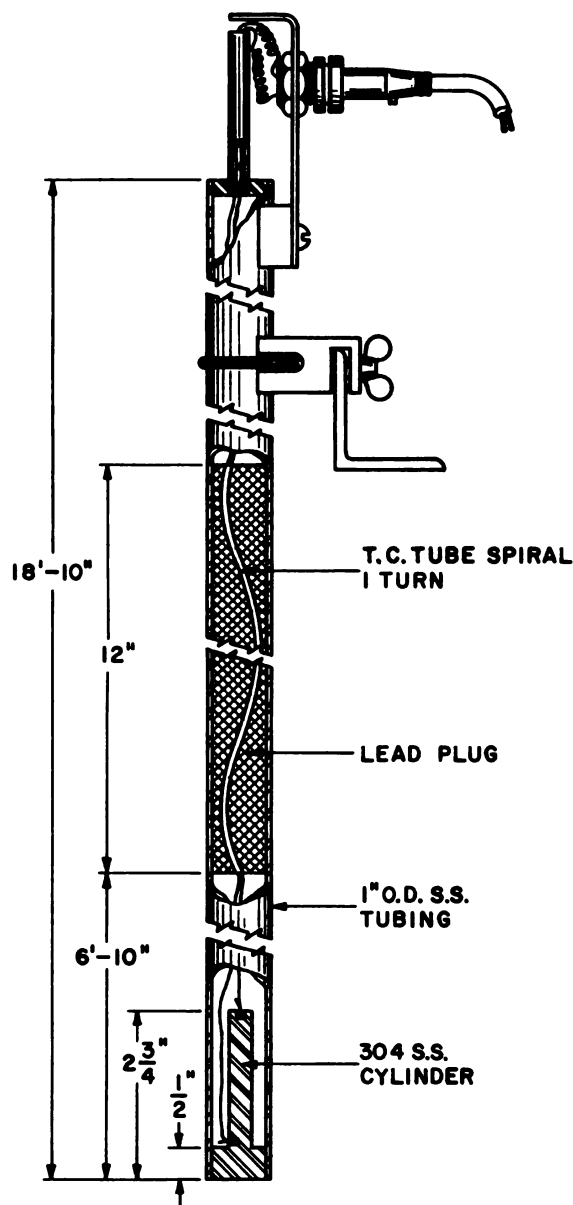


Fig. 3-56 Gamma-intensity measuring apparatus.

the MTR and appear to be satisfactory for use in the range 8×10^3 r to 2×10^6 r. Very often, however, interest is in dosages considerably higher (such as 10^9 or 10^{10} r), and for this range no dosim-

eter is yet available. Thus the intensities must be measured and averaged over the exposure period to obtain dosage.

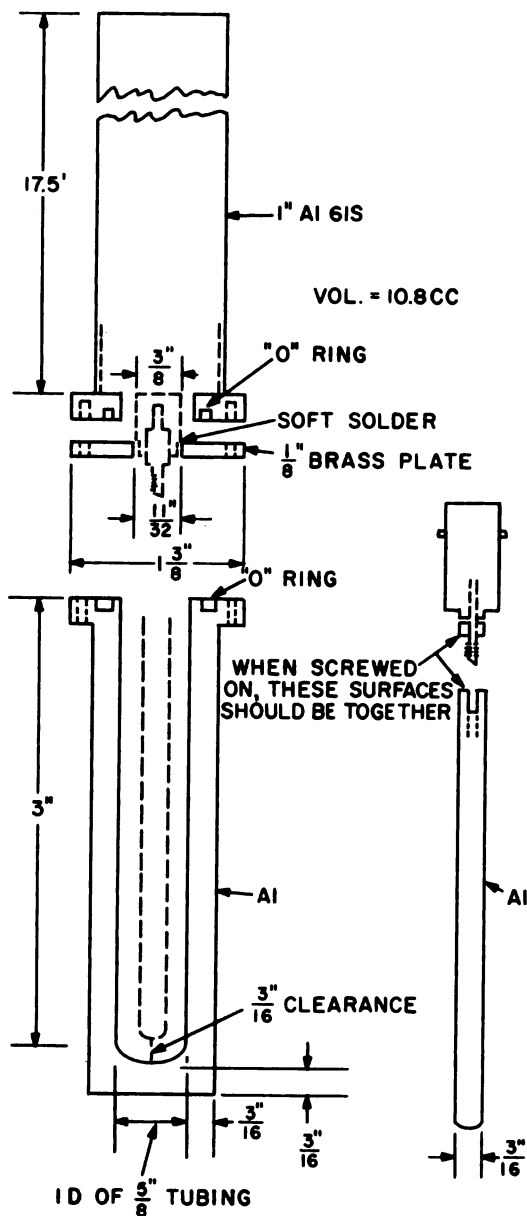


Fig. 3-57 Ionization chamber.

3-76 Applications for the MTR Gamma Facility. Although by no means complete, the following list illustrates the type of materials which have been irradiated in the facility:

1. Electrical insulation.

2. Glass.
3. Polyethylene.
4. Lubricants.
5. Petroleum catalysts.
6. Hydrocarbons.
7. Canned foods—meats, vegetables, milk, etc.
8. Dosimeters, for calibration.

3-77 Disposal of Spent Fuel Elements. It is intended that spent fuel elements from the reactor will be retained either in the gamma grid described above or in similar facilities in the gamma-irradiation building (described in the following section) for a cooling period. After this time the elements are considered decayed to a point where they are not of high enough intensity to be of interest as a gamma source and yet have cooled sufficiently to be processed for recovery of the uranium. Hydraulically operated sawing equipment is then used to prepare the assembly for transfer to the chemical processing plant. This operation calls for the lowering of a carrier through an opening above the canal ceiling into a loading well in the canal floor. The various assemblies ready for transfer are loaded manually into the carrier in a vertical position. After four assemblies have been placed in the carrier or coffin, the cover is put in position and the carrier is lifted by a 20-ton motor crane onto a truck. No provision is made for water circulation in the carrier, since it is believed that thermal siphoning will be sufficient to cool the assemblies. A universal sized carrier for all assemblies appears practical as the sawed length of fissionable assemblies is the same.

3-78 Gamma Irradiation Building Facilities. A gamma-ray facility is being constructed in the MTR area for the use of experimenters. Details of the 2300-ft² building and 6- by 40-ft canal which will house the irradiation facilities are given in Appendix B. The services of the MTR gamma facility will largely be duplicated in this facility. It is expected that the new facility will fill the needs for an unclassified facility for industrial experimenters and university research groups.

Spent MTR fuel elements will furnish the radiation source, thus making the facility a lower radiation source than the MTR facilities but still of substantial gamma intensity when compared to

a Co^{60} source. Initially two types of grids will be available for holding the fuel elements. A universal (checkerboard) type of grid is shown in Appendix B. Thirty-five cadmium-lined boxes will contain the fuel elements. These fuel-element boxes may then be placed in any of 170 sockets in the grid to provide a wide range of configuration with traverse spacing variable from less than 1 in. to 24 in.

A second type of grid is illustrated in Appendix B. Fourteen containers, rotating about their vertical axis, are located on both sides of 10 MTR spent fuel elements. This design affords a uniform

exposure to the material being irradiated and should be particularly useful in food irradiations.

THE MATERIALS-TESTING REACTOR HOT CELL

3-79 This hot cell can handle 10,000 to 20,000 curies of 1.5- to 3.0-Mev gamma activity. It is versatile and can accommodate heavy, bulky equipment or delicate, fragile objects for testing and examinations. The pumice block building housing the cell provides working and storage areas. The cell is built of barytes concrete, the

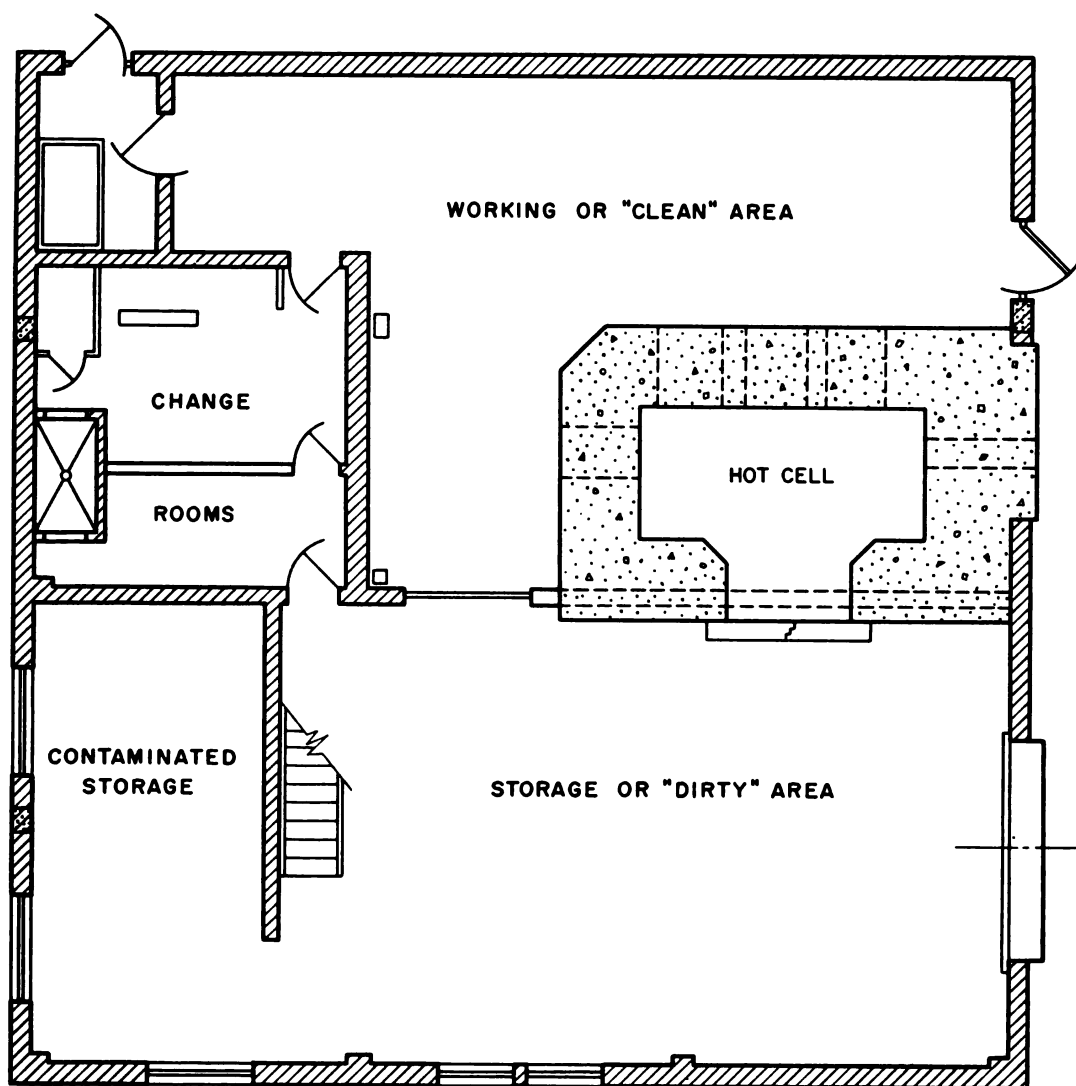


Fig. 3-58 Plan of cell building.

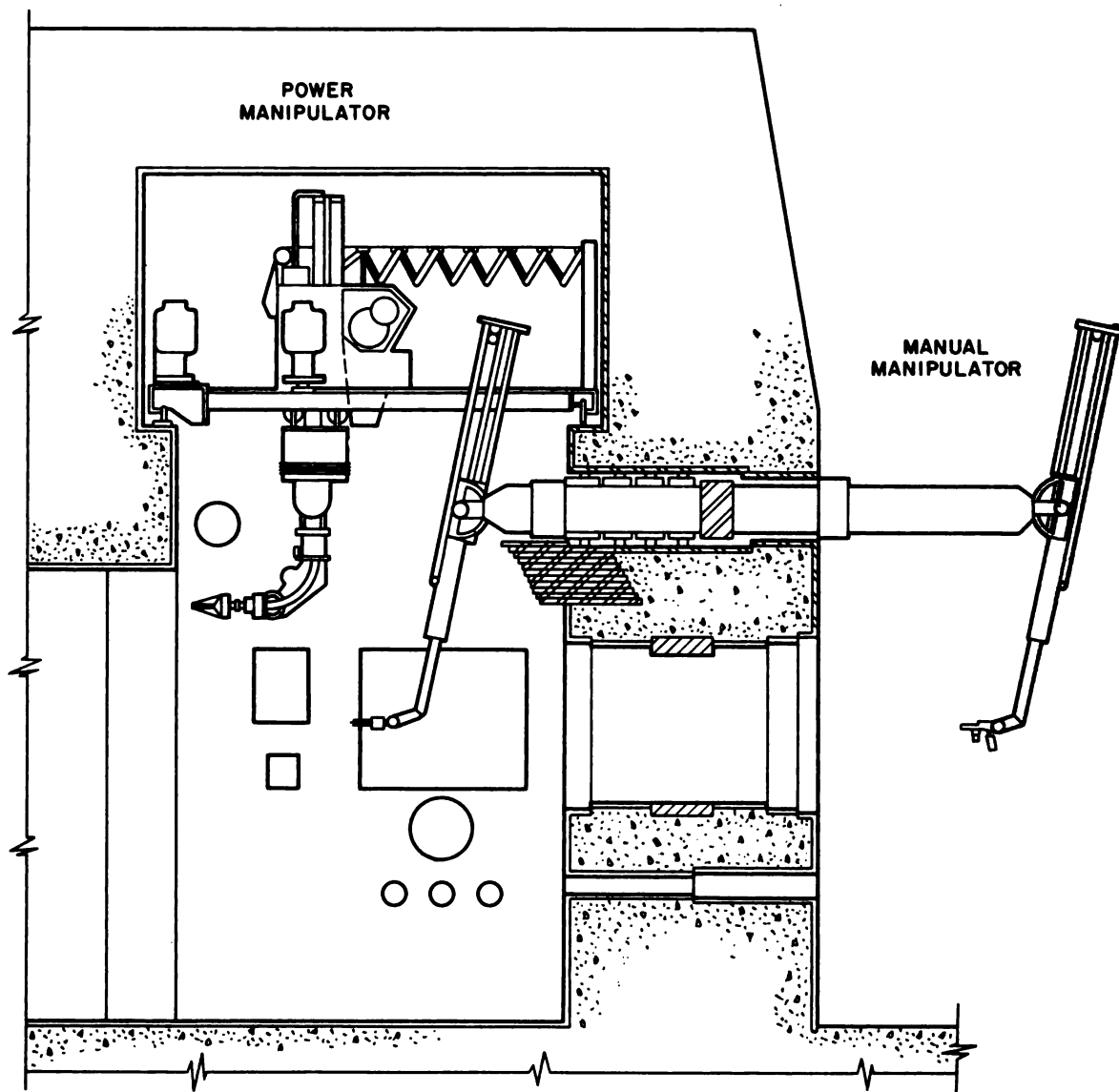


Fig. 3-59 Cross section of hot cell.

walls being 4 ft thick, lined with $\frac{1}{4}$ -in. steel surfaced with protective coatings; the interior measures 14 by 6½ by 13 ft high. Three windows 30 by 36 in. and a smaller end window give maximum visibility and afford the same shielding as the walls through use of 32 in. of zinc bromide solution, 8 in. of high-density glass, and non-browning glass. A General Mills manipulator, a pair of Argonne "master slaves," and a light crane provide flexible remote handling. The exhaust system removes up to 1700 cfm of air, and a separate system exhausts hooded boxes used to han-

dle volatile or dusty substances. Sliding steel doors 18 in. thick close the 6- by 7-ft rear opening. The cell and building were constructed and equipped with windows and manipulators for a total cost of about \$268,000. Plan and section views of the cell are shown in Figs. 3-58 and 3-59.

REACTIVITY MEASUREMENT FACILITY (RMF)

3-80 Beginning with the first pile in the west stands at the University of Chicago, reactors

have been used as neutron sources and as analytical tools. The requirements of reactors as sources conflict with the requirements of reactors as detectors. A detector reactor must fulfill two requirements. It must have high sensitivity to reactivity changes, and its responses must be capable of interpretation. Source reactors fail to meet these requirements because they are large, which decreases their sensitivity, or because operational effects such as the temperature change, xenon growth, or fuel depletion produce such large and rapid reactivity changes that it is impossible to perform or interpret experiments. Thus, reactors destined to be neutron sources, such as the MTR, are rather poor measuring instruments; and reactors which are primarily tools are not the best neutron sources. The fullest exploitation of reactor potentialities is realized by building both source reactors and detector reactors. The fullest use of the combination of a source reactor and a detector reactor is obtained by making it possible to transfer test samples as quickly and cheaply as possible from the source to the detector. This is accomplished at the MTR by placing the RMF in the MTR canal.

The reactivity measurement facility is a small critical facility located in the east end of the MTR service canal. The purpose of this facility is to permit the measurement of reactivity changes in reactor-core materials, temperature-coefficient measurements, and certain reactor physics investigations, such as the evaluation of the gross fission-product cross section as a function of operating time in a high-flux reactor. The core of the RMF consists of an array of standard MTR fuel assemblies, water moderated and water reflected in much the same manner as the Oak Ridge pool. The unique features of this reactor are:

1. The achievement of vertical symmetry by the use of control rods which are either completely removed from the core during operation or are so

arranged that they do not perturb the vertical flux distribution.

2. The use of an optimal spacing between elements so that k changes resulting from element vibration or motion are all negative. This adds considerably to the safety of the design against collapse or destruction of the core by accident.

3. The inclusion of a central thermal column formed by the omission of four fuel elements from the geometrical center of the reactor, leaving thereby a water-filled space approximately 7 by 7 in. This central facility permits some discrimination between k changes arising from fuel and poisons. This can be accomplished because of the low probability that a thermal neutron will diffuse into the center of the thermal column and out again without suffering capture in the moderator material; whereas, if fuel is placed in the center of the thermal column, there is a somewhat higher probability that the fission neutrons resulting from fission will find their way back into the main reactor core.

4. The design of the neutron-indicating elements so to permit the measurements of k changes in the presence of high gamma radiation fields, such as a "hot" fuel sample measured a short time after irradiation in the MTR.

The RMF was loaded to criticality on Feb. 11, 1955, and since that time there has been a program of investigation of the physical constants of the reactor, itself, such as statistical weights, flux distribution, and control-rod effectiveness. Even without the use of oscillator techniques, very small samples can be used for many experiments (milligram quantities of fuel and 0.1 cm^2 of absorber). Present plans include the addition of a pile oscillator and hydraulic and pneumatic rabbit facilities directly from the MTR into the RMF. The addition of these facilities will greatly enhance the present utility and sensitivity of the reactor.

PART 6 ENGINEERING TESTING

INTRODUCTION

3-81 The Materials-testing Reactor was designed primarily to test various materials in high-intensity radiation fields. To permit this, about 100

ing fuel in the reactor lattice in several different patterns. The standard loading adopted for the present is a 3×9 "north slab core" (see Fig. 3-40). This loading has fuel assemblies in all positions of the three northernmost rows of the

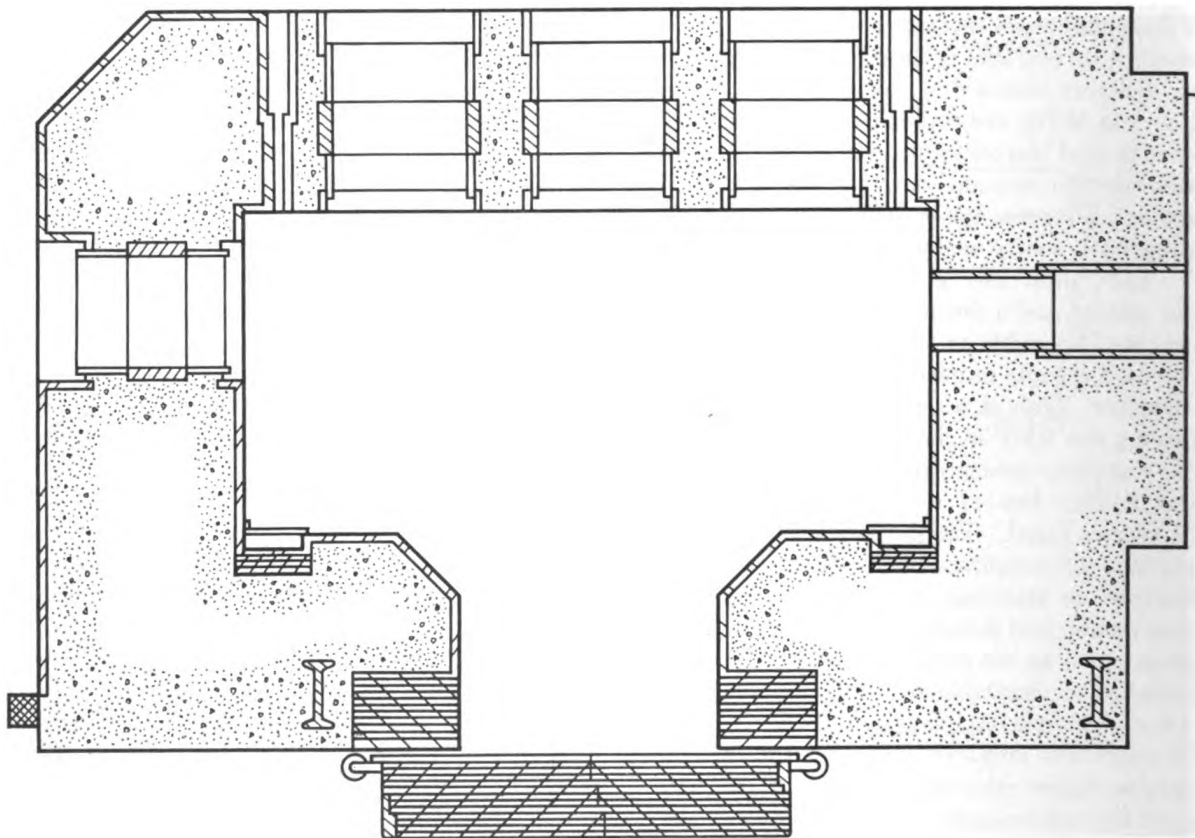


Fig. 3-60 Plan of hot cell.

experimental holes were provided through each of the four walls and both the top and bottom of the reactor.

Facilities for experimental work inside the reactor tank are divided into four groups:

1. Reactor-tank experimental access holes.
2. Reactor-lattice facilities.
3. Reflector facilities.
4. Hydraulic facilities.

The MTR active lattice can be formed by load-

ing fuel in the reactor lattice in several different patterns. The standard loading adopted for the present is a 3×9 "north slab core" (see Fig. 3-40). This loading has fuel assemblies in all positions of the three northernmost rows of the reactor lattice except positions 22, 24, 26, and 28, which are occupied by cadmium-uranium shim rods. The remaining two rows are normally occupied by beryllium L pieces except that positions 42, 44, 46, and 48 contain beryllium-cadmium shim rods. The following discussion assumes this standard loading, and the neutron fluxes are "unperturbed" values, i.e., they do not take into account the effects due to experiments in the reactor.

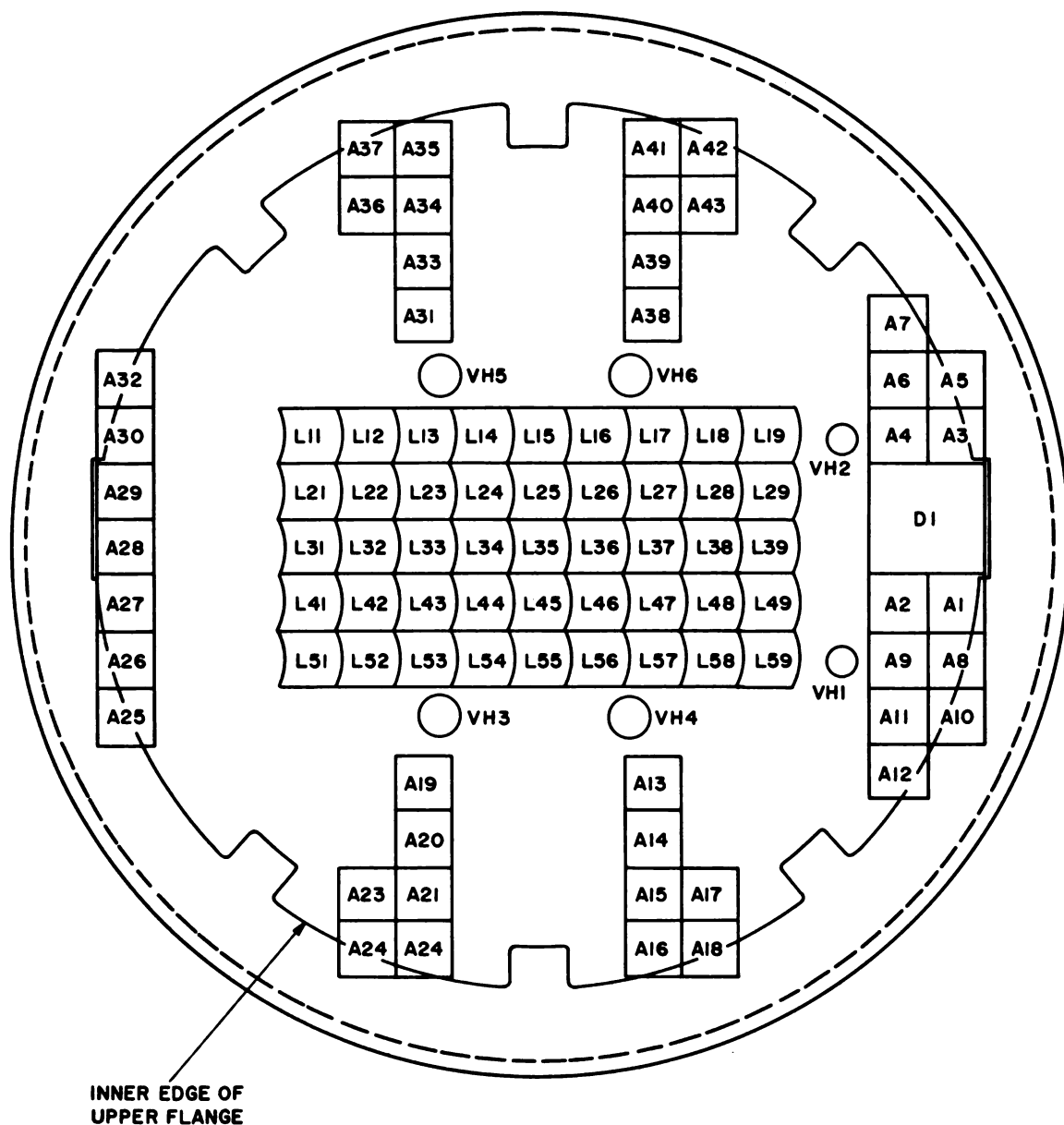


Fig. 3-61 Plan of reactor lattice and beryllium reflector.

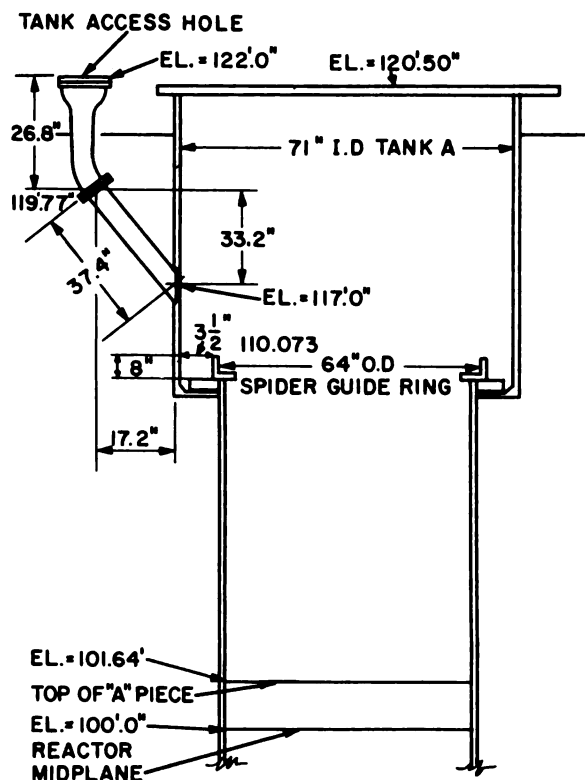


Fig. 3-62 Cross section of tank with elevations.

FACILITIES FOR IRRADIATION WITHIN THE TANK

3-82 Reactor-tank Experimental Access Holes. Two access holes to the reactor tank were added to the reactor late in its construction. These provide access to the reactor tank through the process-water inlet lines on the north and south. The location of these holes is shown in Fig. 3-62, which shows a cross section of the reactor tank through one of the access holes. The elevations are relative to the reactor horizontal center line at 100 ft.

Electrical leads, conduits, etc., to an experiment in the reactor tank are through a blind flange bolted to the top of the access-hole flange. The leads must have watertight seals capable of withstanding 75 psi water at the blind flange. Only two access holes are available for all experiments in the reactor tank; consequently each experimenter should design his leads and seals so as to use a minimum of flange area. The velocity of the inlet water causes vibration of the leads within the tank, and therefore clamping devices must be

used inside the tank. Leads or conduits must have some flexibility because the access holes enter the reactor tank at an angle.

3-83 Reactor-lattice Facilities—L Pieces. The framework that normally contains the fuel, shim rods, and beryllium pieces is called the reactor lattice. A row-column number system is used in designating the lattice positions (see Fig. 3-61). With this system the shim-rod positions are L-42, 44, 46, 48, 22, 24, 26, and 28.

Soon after startup of the reactor the beryllium L pieces in the reactor were replaced by modified L pieces, referred to as LB pieces. These pieces have a $1\frac{3}{8}$ -in. hole through the beryllium approximately 37 in. long. Cylindrical capsules $1\frac{1}{4}$ in. in diameter may be inserted in the LB pieces with a cooling-water annulus $\frac{1}{16}$ in. wide around them. The capsule length depends on the needs of the experimenter. The capsule must be centered in the hole to assure adequate cooling on all sides. Figure 3-63 shows the dimensions of an LB piece. There are 14 LB pieces available at the present time.

Aluminum pieces similar in design to the LB pieces are also available. These pieces are made of 2S aluminum and are designated as LA pieces. One advantage of the use of an LA piece is that samples irradiated in them in row 4 of the reactor lattice, which is adjacent to the active core, will be subjected to higher fast flux than if LB pieces were used (Fig. 3-64).

Shim-rod hole. The shim-rod position L-42 has been modified to provide access through the subpile room. The position contains a dummy shim rod with a hole throughout its full length that will accommodate a thimble of 2 in. OD.

3-84 Reflector Facilities. The beryllium reflector occupies the region between the reactor lattice and the tank wall. The pieces normally used for experimental purposes are the A positions numbered 1 to 43 (see Fig. 3-61). The beryllium-reflector pieces used for experiments are designated AB pieces and provide for the irradiation of standard-sized capsules or plate samples formed into rings. The pieces are about 3 by 3 by $38\frac{3}{4}$ in. long. The piece with the single hole provides a space $1\frac{3}{8}$ in. in diameter and $38\frac{3}{4}$ in. deep. Figure 3-65 shows the construc-

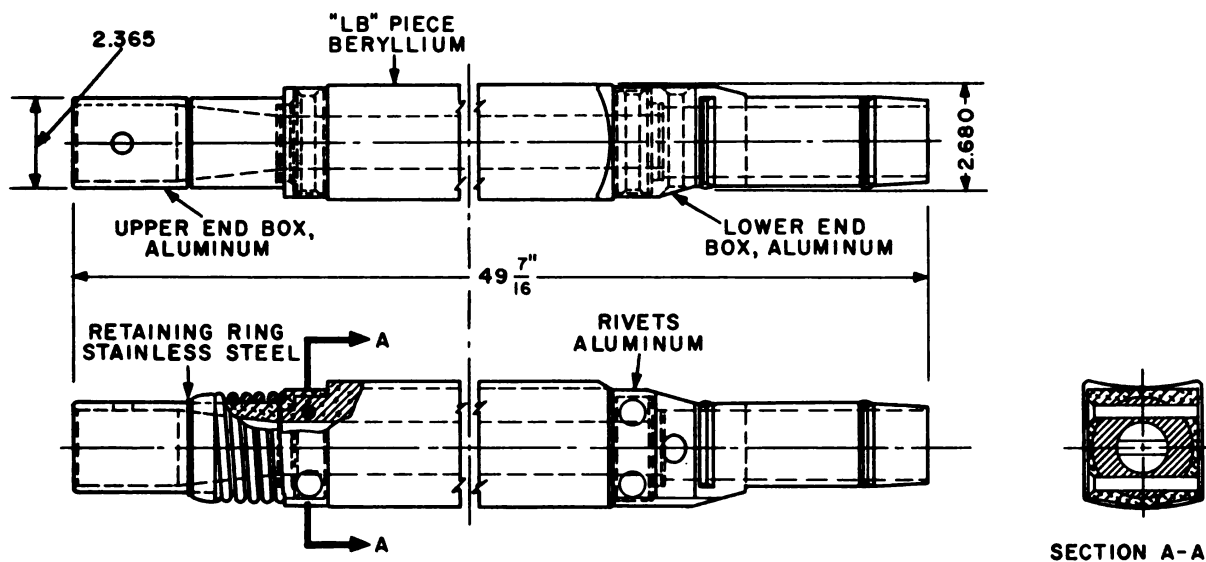


Fig. 3-63 LB piece (beryllium).

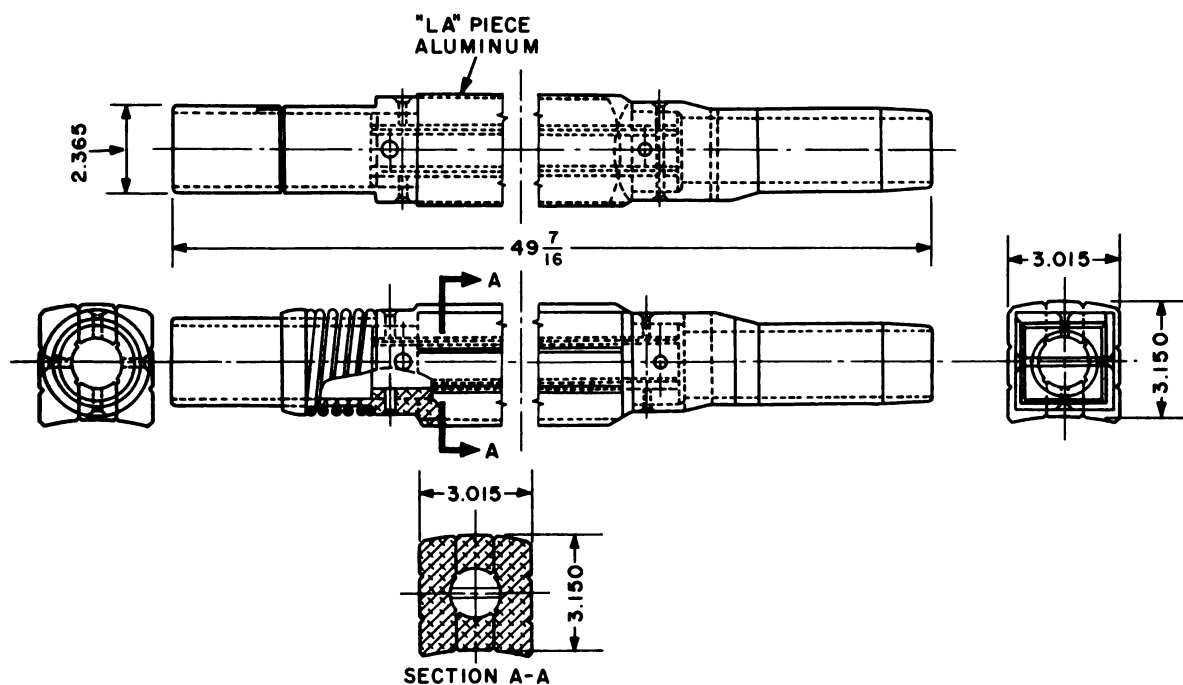


Fig. 3-64 LA piece (aluminum).

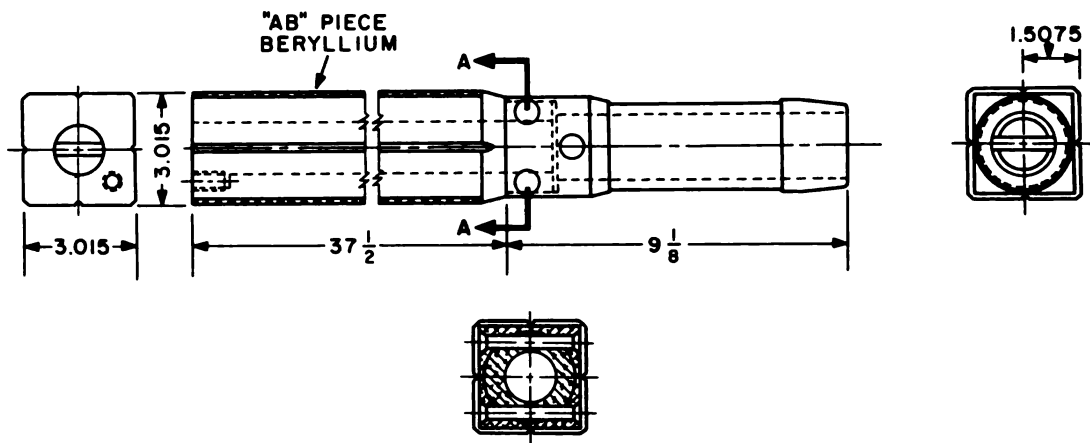


Fig. 3-65 AB piece (beryllium).

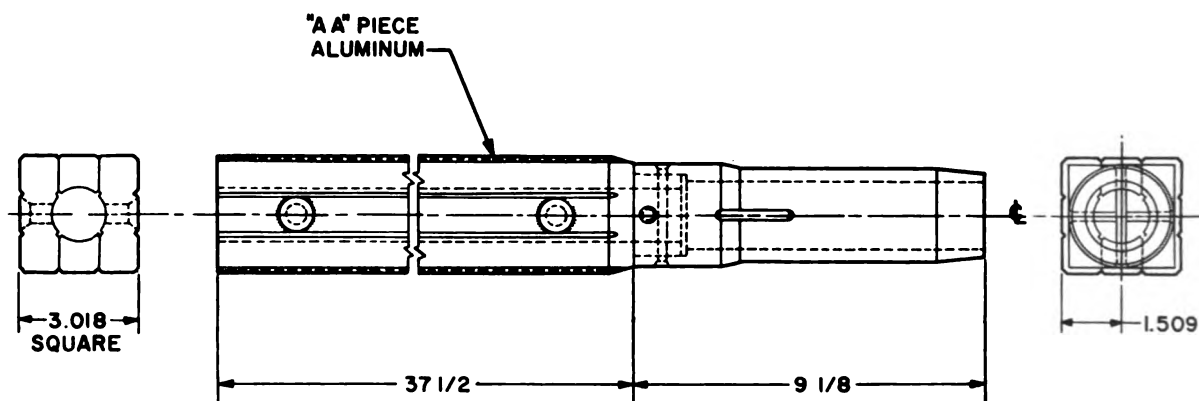
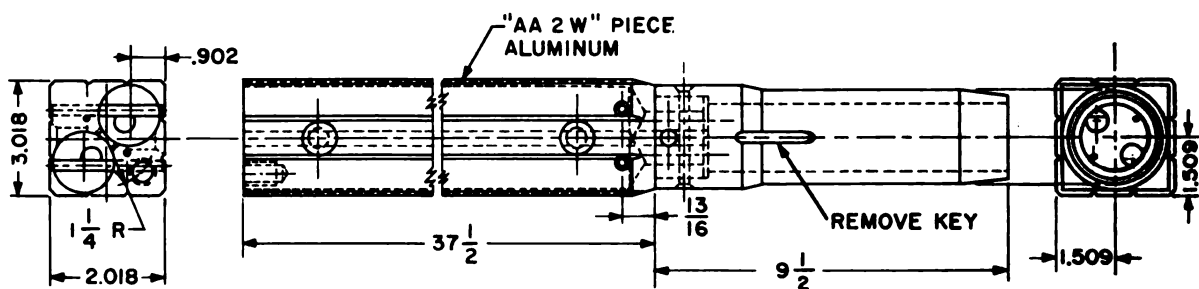


Fig. 3-66 AA piece (aluminum).



NOTE -

THIS ASSEMBLY IS A MODIFICATION OF
EXISTING ALUMINUM "A" PIECES.

Fig. 3-67 AA-2W piece.

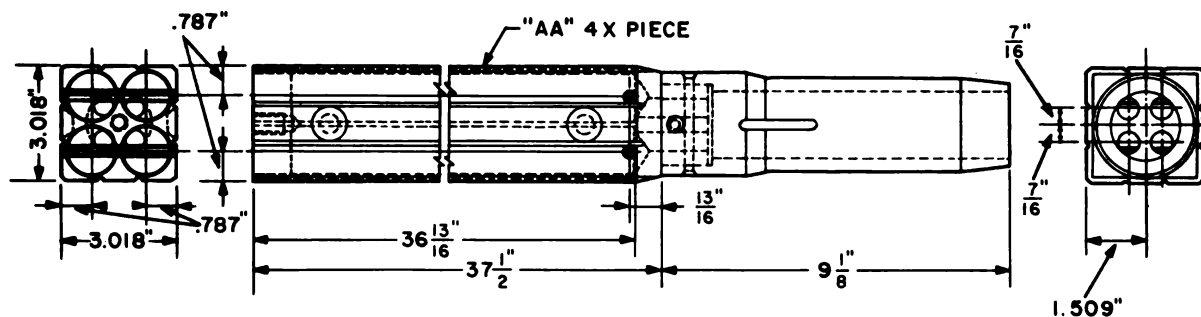
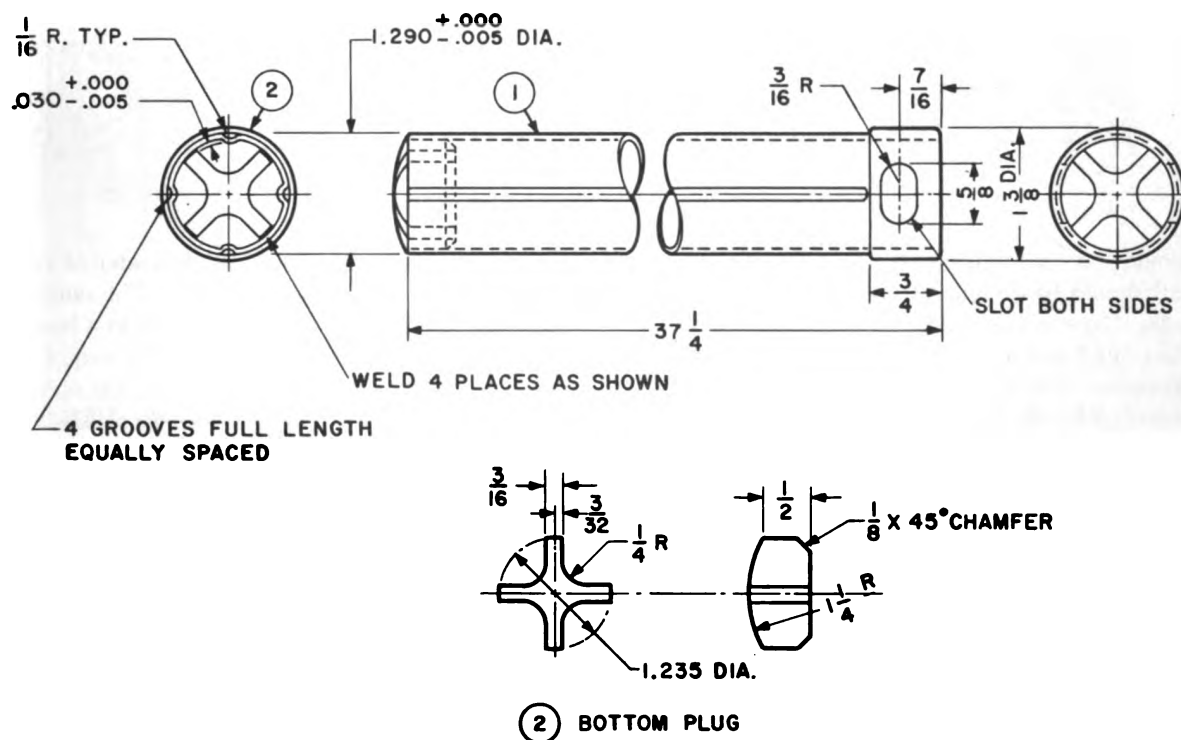


Fig. 3-68 AA-4X piece.

tion of an AB piece. The corresponding aluminum piece made from 2S aluminum is designated AA and is shown in Fig. 3-66. In addition to the single-hole piece, aluminum pieces containing two

and four holes are available for irradiation of samples.

Experimenters should design experiments to fit the two- or four-hole pieces since space in the re-



**NOTE: TUBING I.D. TO BE 1.244" TO 1.259".
I.D. TO BE FREE OF BURRS.**

Fig. 3-69 X basket for AA-4X piece.

actor is at a premium. The two-hole pieces (AA-2W pieces shown in Fig. 3-67) provide two holes 1.533 in. in diameter and $36\frac{13}{16}$ in. long, while the four-hole or AA-4X pieces (Fig. 3-68) provide four holes 1.312 in. in diameter and $36\frac{13}{16}$ in. long. Capsules irradiated in these facilities

through a bend of 4 ft minimum radius in the hydraulic tubes and must allow adequate water to flow past the sample to provide cooling. The sample must have an effective density greater than that of water. Figure 3-70 shows a schematic diagram of the hydraulic facilities.

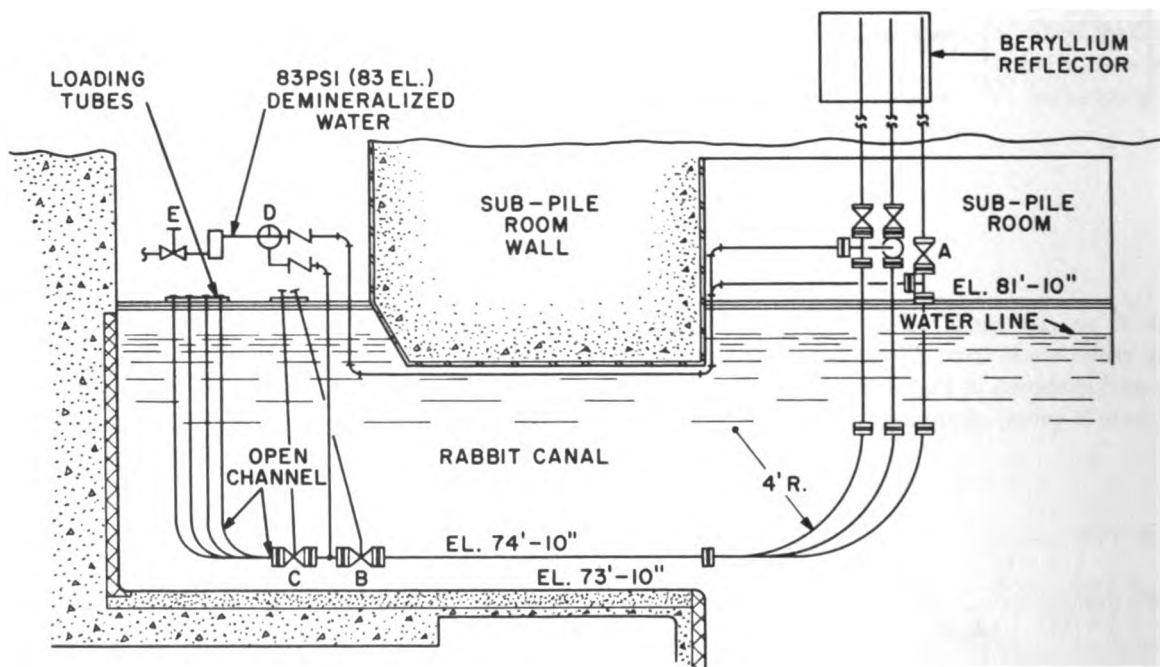


Fig. 3-70 Schematic diagram of hydraulic shuttle facilities.

should have a cooling-water annulus $\frac{1}{16}$ in. wide and should be designed to remain centered in the hole. Capsule baskets are available for the AA-4X pieces and will accommodate a capsule $1\frac{1}{8}$ in. in diameter. Centering guides for the capsules are provided by the basket, as shown in Fig. 3-69.

3-85 Hydraulic Facilities. The MTR hydraulic facilities permit the irradiation of capsules in the high-neutron-flux fields near the reactor lattice for periods ranging from a few minutes upward. These facilities are located in the VH-1, 2, 3, and 4 positions (Fig. 3-61).

The VH-1, VH-2 tubes are 1.000 ± 0.005 in. ID, while VH-3, VH-4 are 1.313 ± 0.005 in. ID. The capsule is forced into the facility hydraulically and comes to a stop at the end of the tube in the reflector. The sample is cooled by means of the demineralized water used to hold it in place in the tank. The sample must be able to pass

Table 3-36 gives the standard diameter of capsules for use in the various facilities. The samples for hydraulic facilities are usually kept to a length of $3\frac{1}{2}$ in.

Table 3-36 Sizes of Various Facilities—MTR

Facility	Facility diameter, in.	Sample diameter
Napkin-ring stringer baskets..	0.650	$\frac{5}{8}$
Hydraulic rabbits, VH-1, 2...	0.995	$1\frac{3}{16}$
Hydraulic rabbits, VH-3, 4...	1.310	$1\frac{1}{8}$
X baskets.....	1.19	$1\frac{11}{16}$
Four-hole A pieces.....	1.312	$1\frac{1}{4}$
One-hole A pieces.....	$1\frac{3}{8}$	$1\frac{1}{2}$
W baskets.....	1.453	$1\frac{3}{8}$
Two-hole A piece.....	1.553	$1\frac{11}{16}$
One-hole A piece.....	$2\frac{1}{16}$	$2\frac{1}{2}$
AA-ANL-1a, 2a, 3a.....	$1\frac{1}{8}, \frac{3}{8}, \frac{5}{8} (2)$	

NEUTRON FLUXES

3-86 The thermal-neutron-flux pattern for the 30-megawatt power level and the horizontal mid-plane of the reactor is shown in Fig. 3-71. This "map" takes into consideration, to some extent, the effect of experimental holes. Depending on the nature and number of the experiments in the reactor, actual neutron fluxes encountered will be somewhat less than those shown.

Six down-beam holes, available from the balconies on the north and south, go to the reactor tank and "see" the lattice through 2-in. holes in the beryllium reflector (Fig. 3-3).

Figure 3-22 shows the layout of a typical horizontal beam hole and its shielding plug. The parts of plugs are made of beryllium where they reside in the beryllium reflector, of graphite in the reactor graphite, of iron through the thermal shield, and of concrete in the biological shield.

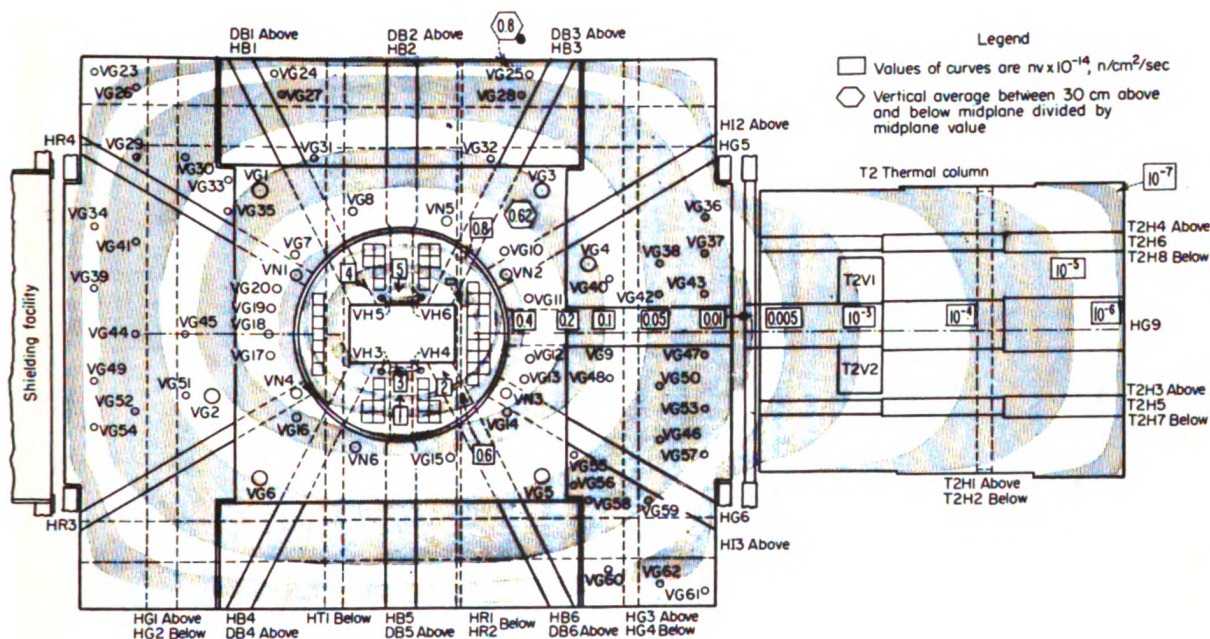


Fig. 3-71 Thermal-neutron-flux pattern at reactor mid-plane (30,000 kw).

Figure 3-73 is a map of the fast flux in the reactor at the same location and under the same conditions as above.

EXPERIMENTAL FACILITIES OUTSIDE THE REACTOR TANK

3-87 Six horizontal beam holes and a through facility allow experiments to be performed within 2 in. of the active lattice. Figure 3-4 shows the location of these holes. They are accessible at the outer face of the concrete shield. Since these holes penetrate to the highest flux in the reactor, they are potentially hazardous from a radiation point of view, and their use is therefore different and complicated. Each must be filled with a shielding plug during reactor operation.

For experimental work, a new plug must be designed and built with the experimental equipment attached to the front end or embedded in the plug, and properly cooled. Instrument leads, cooling connections, etc., can be brought outside.

The plugs can be removed only during shutdown of the reactor, and even then residual radiation is such that special radiation doors in the beam holes must be closed. Since the portions of any plug inside the thermal shield become highly radioactive, heavy lead-shielded casks are required for handling them. A so-called "universal coffin" is provided to handle the horizontal beam-hole plugs. Ten inches of lead in the front end of the coffin shields the beryllium parts of these plugs. The gross weight is 41,000 lb. The plugs are transported in the coffin to a plug-storage facility

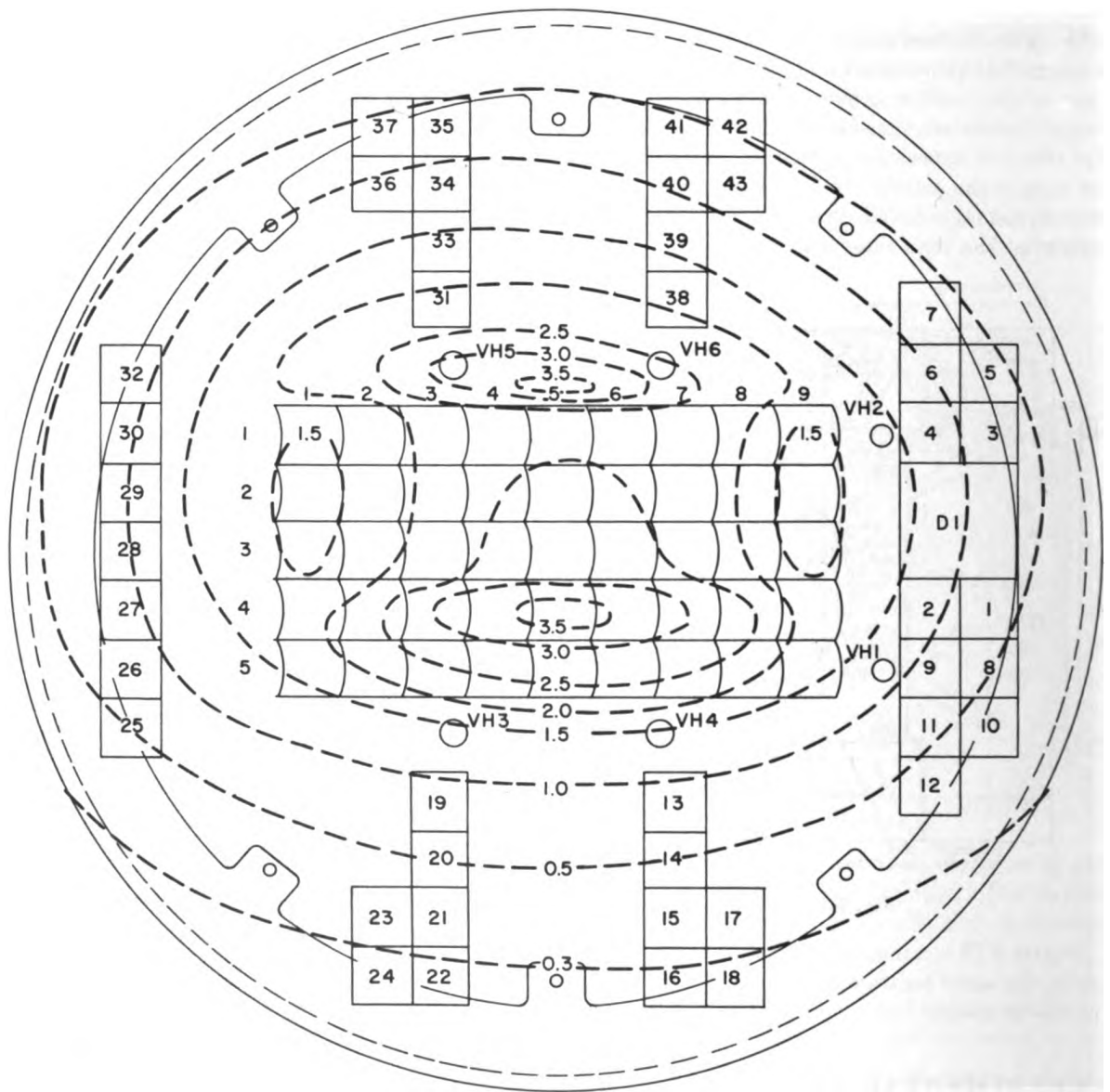


Fig. 3-72 Reactor cross section showing maximum flux values for charge 36.

which is located outside the Reactor Building where they are stored until they are needed again.

In addition to these main experimental holes, there are six more which run horizontally through the graphite wall. Four of these holes are for experimental work, and two are for reactor-control instruments. They are either 4 or 8 in. in diameter.

VERTICAL EXPERIMENTAL AND INSTRUMENT HOLES

3-83 Much irradiation space at reduced neutron fluxes is available in holes accessible from the top of the reactor. Some of these penetrate into the pebble graphite and some into the permanent graphite; two are also accessible from a room under the reactor. The dimensions range generally

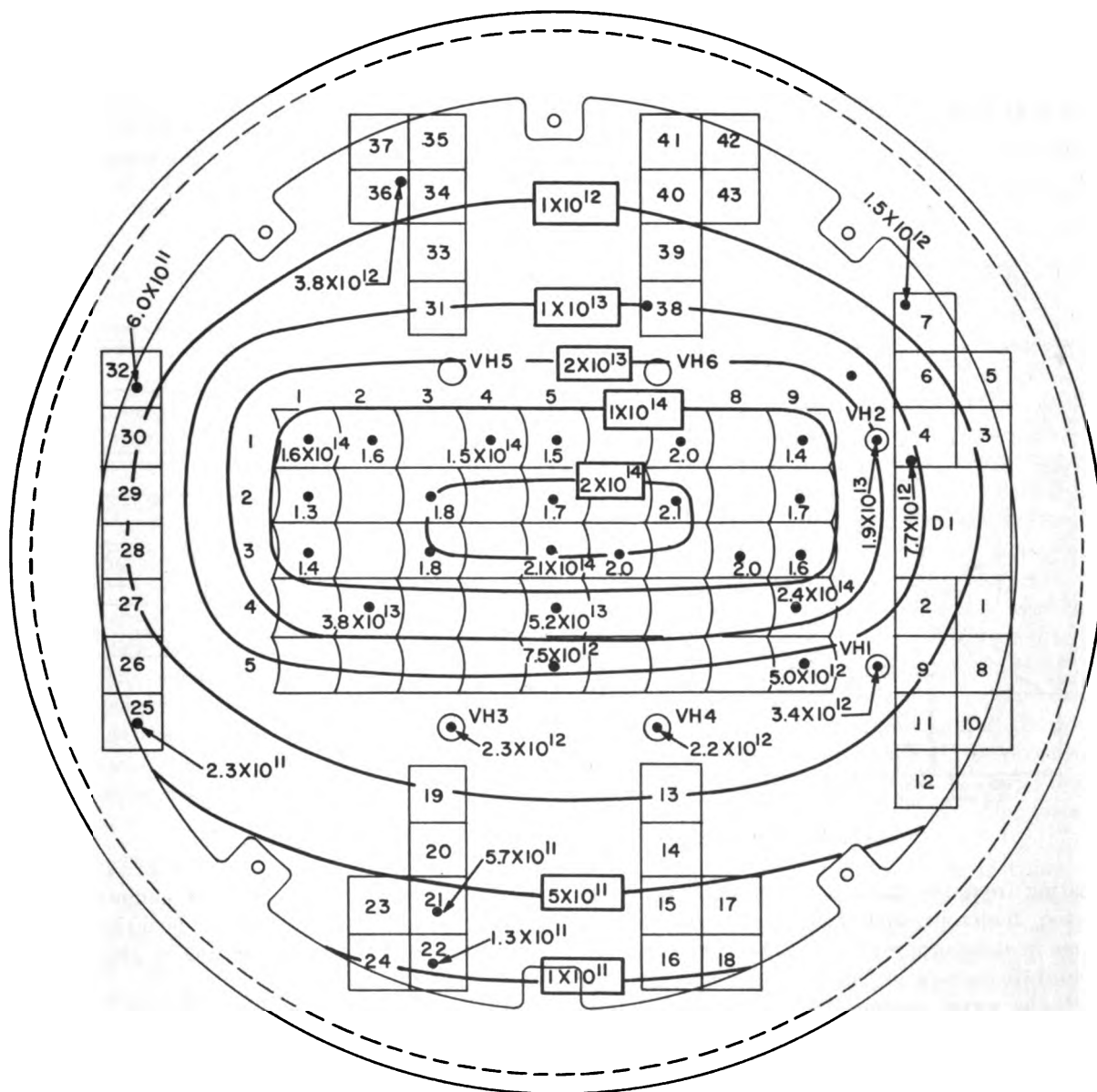


Fig. 3-73 Preliminary map of fast-neutron flux at reactor mid-plane.

from 2 to 4 in. in diameter. There is also space in four 6-in. thimbles which extend into the 30-in. cooling water pipes. In two of these thimbles are instruments for measuring the radioactivity of the exit water. A high-gamma neutron-free flux is available for experimental purposes in the other two. Figure 3-4 gives the location of typical VG holes.

THERMAL COLUMN

3-89 A "thermal column," 6 ft square and 8 ft long, built up of graphite blocks, extends from the east face of the permanent graphite wall to provide a very pure thermal-neutron flux. Several experimental holes penetrate it. Four inches of lead at the inner face cuts the gamma-ray flux in the column and, with an additional 10 in. at

the outer face, reduces the radiation from the reactor to a harmless level. A cross section of the thermal column is shown in Figs. 3-4 and 3-5.

GAMMA-RAY HEATING

3-90 Most materials placed in the reactor tank or graphite reflector are subject to considerable

can be handled for almost any material being irradiated. In the graphite, the gamma heating is significantly lower than in the tank, but because air cooling only is provided, heat removal must be analyzed carefully. Superimposed on this gamma heating from the reactor is any fission or capture heat developed by the material being irradiated (see Fig. 3-74).

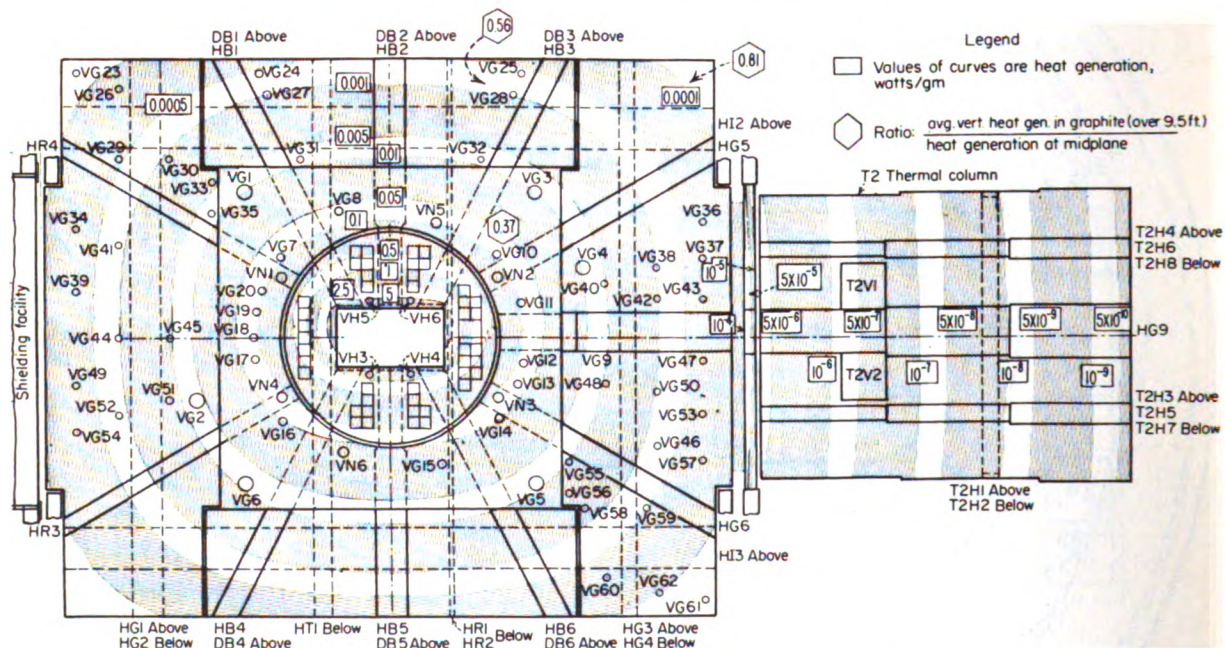


Fig. 3-74 Heat-generation pattern at reactor mid-plane (30,000 kw).

heating from the gamma rays which arise from fission, fission-product decay, and neutron captures in structural material. The values go from 5 to 0.10 watts/g in the beryllium reflector, but with the water cooling available there this heat

Table 3-37 gives the values of gamma heating to be expected in some typical facilities in locations where the gamma heating is greater than 1 watt/g.

Experiments which use the horizontal beam

Table 3-37

Location	Heat, watts/g	Location	Heat, watts/g
HB-1, 2, 3	10	A-28, 29, 30	1.0
HB-4, 5, 6	2	A-31, 38	2.5
L-51 to 59	14	A-33	1.5
L-41, 42, 43, 45, 47, 49	14	A-39	1.25
A-1	0.75	HG-9	2.5
A-2	1.5	VH-1, 2, 3, 4	1.5
A-4	2	HT-1	2.5

holes are somewhat complicated to design and perform. Construction of the reactor requires that such an experiment pass through the concrete biological shield, the permanent graphite zone, the pebble graphite zone, and finally the beryllium reflector. Experiments most conveniently performed in beam holes are those involving high-temperature high-pressure water loops or high-temperature liquid metal loops. In order to perform such experiments, careful attention must be given to the heat loss to the reactor structure. Table 3-38 gives the allowable values for

Table 3-38 Allowable Heat Losses to Reactor Structure

Be reflector . . .	3.6 kw/in. of length
Pebble zone . .	3000 Btu/hr/ft of length
Graphite	6000 Btu/hr/ft of length
Concrete * . . .	600 Btu/hr/ft of length

* Without auxiliary cooling coils.

heat loss in the various sections along the beam hole.

Beam-hole experiments are also complicated by the fact that the experimental plug, about 11 ft long and 6 in. in diameter, must contain experimental components, leads, and sufficient shielding to reduce radiation to safe biological levels outside the reactor. The irradiated experiment must be handled using a 41,000-lb coffin.

EXPERIMENTS

3-91 A few problems associated with the insertion of a typical experiment in the MTR will be considered here. The example will be a high-pressure high-temperature water loop to be designed for one of the beam holes; the experimental purpose of such a loop could be to study the corrosion of fuel material under irradiation, the effects of irradiation and corrosive environment on cladding materials, the effects of irradiation and corrosive environment on the loop materials, or chemical studies on the water. The loop should withstand the stresses necessary to keep the pressure high enough to maintain the surface temperature of all in-pile components at least 35°F below the water-saturation temperature. It should be possible to insert and discharge the experimental element from the in-pile thimble without removing the loop. All pressure vessels

should conform to the standards of the ASME Code for Unfired Pressure Vessels, and piping should conform to the American Standard Code for Pressure Piping. The experimental setup will consist of a primary coolant system, a water-purification system, and a water make-up system (external to the reactor), and an in-pile test thimble. Also external to the reactor are the necessary control and supervisory circuits necessary to operate the experiment. Figure 3-75 shows a schematic flow diagram of a typical loop experiment. At least two primary circulating pumps should be provided with the system, one of which is for emergency use only in case of failure of the normally operating pump. Check valves prevent circulation through the stationary pump. High-pressure water is circulated through the system to remove fission and gamma heat from the in-pile section. The circulating water is cooled by a heat exchanger. In general, the cooler should have a greater capacity than the heat generation rate in-pile, the overcooling being compensated for by temperature-controlled heaters on the inlet side of the thimble. Pressure is maintained by a pressuring "boiler" or surge tank connected to the loop and operated at saturation by pressure-controlled electric heaters. Some means of overpressure relief for the surge tank should be provided.

The water make-up system should supply pure degassed water while in operation. The system uses high-pressure feed pumps.

In normal operation of the loop, instruments indicate experimental conditions of flow, pressure, temperature, conductivity, surge-tank level, and water activity. Many of the instruments must of necessity be connected to the reactor-control circuit to initiate reactor shutdown when adverse conditions reach a predetermined level. Conditions which could initiate reactor shutdown are (1) low pressure, (2) high pressure, (3) low flow, (4) high temperature, (5) low surge-tank level.

The nature of the reactor shutdown will depend on the urgency of the situation. The types of power reductions will be discussed later. All instruments detecting the conditions just listed should be supplied with audio and visual alarms.

To ensure circulation at all times, an emergency power supply should be provided to supply power automatically if normal power is interrupted.

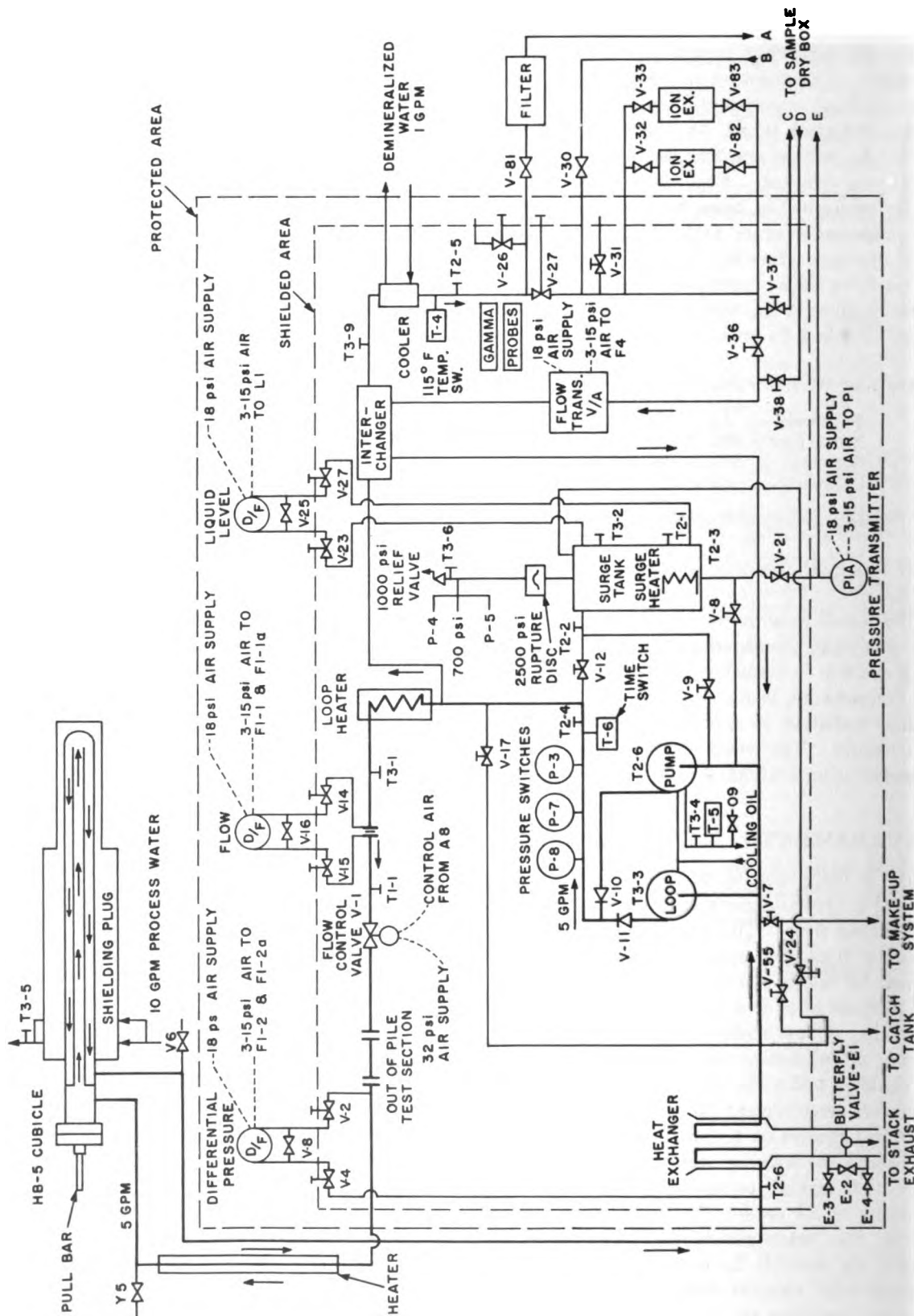


Fig. 3-75 Schematic flow diagram.

Other operations which should be given consideration are charging and discharging the specimen and decontamination of the loop in case of fission-product contamination.

REACTOR CONTROL AND SAFETY CIRCUITS

3-92 The information contained on the following pages is based on experience in the safe operation of experiments at the MTR. The general application of the principles outlined will reduce the chance of incidents to a lower probability. Specific experiments may have special problems that are not covered by application of these general rules.

Facilities available to experimenters for reactor control, and the dynamics of reactor power associated with these controls, will be described. An elementary diagram is included to demonstrate the use of these controls (Fig. 3-76).

Most experiments in the MTR from which information is extracted on a continuous basis are connected with the reactor-control system. Usually this connection is to an annunciator peculiar to the experiment. The annunciator signals a warning in the control room that conditions requiring attention have developed in the experiment. Dangerous conditions which can develop in an experiment in a time too short for an operator to take corrective action must produce automatic reactor power reduction.

Permanently wired plug outlets are situated near the various experimental facilities. Annunciators and all rates of power reduction are available at these outlets. Each experiment requiring more than an annunciator must provide some means of releasing the reactor-control system from false signals for power reduction. The release can be in the form of a switch or, in some cases, test blocks. Details of the use of these facilities are given below.

MTR POWER REDUCTIONS

3-93 Five rates of automatic power reduction are available to experimenters. The rate permitted for the individual experiment is the minimum required to prevent damage to the reactor

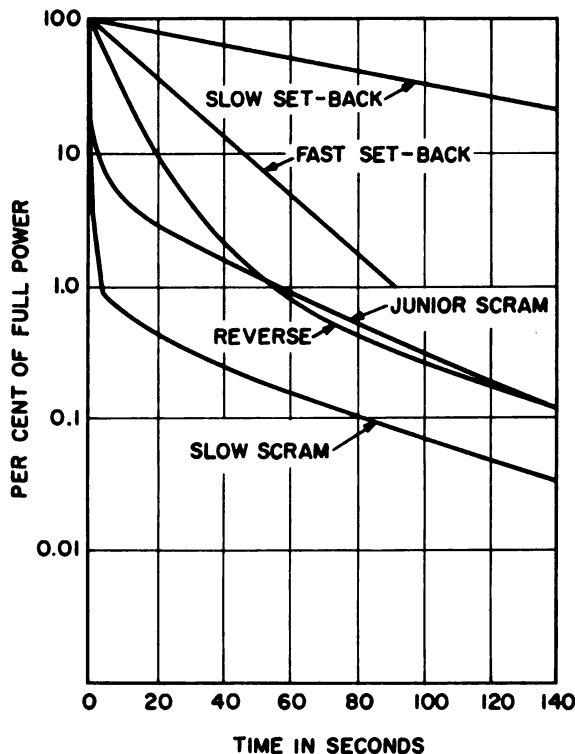


Fig. 3-76 Approximate power vs. time for automatic power reduction.

and the experiment, if trouble develops. Figure 3-76 shows the rates of power reduction available. They are defined as follows:

scram: The release of all seven shim-safety rods which are driven into the lattice by gravity and water pressure.

junior scram: The release of shim-safety rods 6 and 7 which are driven into the lattice by gravity and water pressure.

reverse: A motor-driven insertion of all shim-safety rods.

fast setback: A logarithmic reduction of 1 per cent of full power in 90 sec.

slow setback: A logarithmic reduction of 1 per cent of full power in 420 sec.

In addition to these, nonautomatic power reductions of almost any rate are available to the operator.

It should be pointed out that the slower the rate of power reduction, the more quickly full power can be regained after termination of the call for the decrease. The rates of power reduction are shown graphically in Figs. 3-76 and 3-77. Figure 3-77 also shows the 100-msec delay necessary to operate relays in the control circuit.

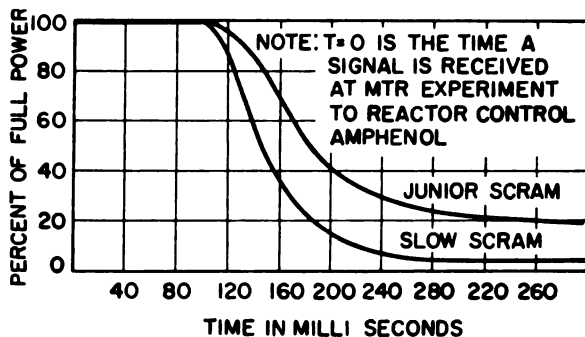


Fig. 3-77 Approximate power vs. time for slow scram and junior scram.

CONTROL REQUIREMENTS

3-94 Engineered experiments in the MTR are generally of such a nature that malfunction of some component will often lead to excessive temperature, pressure, activity, etc., with the possibility of extensive damage or contamination and loss of operating time. This situation dictates the general requirement that protective devices be "backed up," to guarantee that no one channel is relied on entirely to relieve a dangerous situation. This backup principle as applied to reactor and experiment control takes the following general form:

1. A dangerous condition in the experiment is detected by two independent systems, but not necessarily simultaneously. For example, loss of cooling water flow could be backed up by an instrument indicating a loss of pressure or an instrument showing high temperature.

2. A request for scram, junior scram, or reverse operates a relay or pair of relays at the experiment whose contacts pick up two relays in the reactor instrument room. The relays in the instrument room actuate the pair of relays which give the power reduction called for. There are several possible points of failure, and the circuit is duplicated at all points except the one wire supplying uninterrupted power to the experiment. The

availability of uninterrupted power at the experiment is monitored by a separate normally energized relay.

3. A request for fast setback operates a relay at the experiment whose contacts pick up the fast- and slow-setback relays in the reactor-control circuit to initiate the fast and slow setback. The fast-setback circuit is not a double-tracked circuit, and the slow setback is considered to back up in this case.

4. The same sequence of events follows a request for slow setback. This circuit is not double tracked. A backup is not used since the power-reduction rate of the slow setback is over a 7-min period.

The annunciator circuit for the experiment initiating the power reduction must operate before the actual call for power reduction.

Supervisory alarms and lamps should be provided at the experiment, indicating its status at all times.

Sound-powered telephone communication with the reactor-console operator is made available for each experiment.

The purpose and function of the reactor release switch has been described previously. Two other aspects of this switch need to be mentioned. One is the key interlock feature. Whenever the switch is in the OFF position, or the Amphenol plug is disconnected, or there is loss of reactor-control power to the experiment, that experiment's annunciator gives an alarm in the control room. To silence this annunciator when the experiment is not in operation, a similar switch in the south cubicle on top of the reactor is opened, which disconnects the experiment from all connection with reactor control including those circuits not rendered ineffective by the experiment's reactor release switch. To ensure against unauthorized use of the cubicle switch and to make certain that the correct switch is thrown when so authorized, it is made a keylock and can be opened only by using the key from the particular experiment's reactor release switch.

The second aspect of the switch is its use as a device, where needed, to change the character of the experiment's supervision of reactor power under different operating condition. For example, where the experiment running time does not correspond with the reactor-charge cycle, there will

be periods during which the experiment will be withdrawn from the high-flux zone, but all instruments will be in service. A position on the switch will be used to set up the less drastic reactor power supervision required under these operating conditions. An interlock should, of course, be provided to activate the annunciator in case the switch is not in the position appropriate to the operating conditions. If more than two operating conditions are to be supervised, it is more satisfactory to use test blocks in place of the switch.

CIRCUIT DETAILS

3-95 Experiments communicate with reactor control only through relay contacts and the override switch contacts. Power through these contacts is furnished from the reactor uninterrupted power supply and from the same bus which feeds all reactor-control circuits. To avoid a possible cross connection between these and the experiment's power sources, they must not be cabled together, or run in the same conduit, or occupy the same terminal block with wires of the other system. Preferably, all relays associated with reactor control and the bypass switch should be grouped in one place to avoid running reactor-control circuits throughout the experiment.

Neither side of the reactor-control power should be grounded. This system is balanced to ground and monitored for disturbance of this balance. Voltage to ground will not exceed 120 volts rms at the experiment. All wiring should conform to a standard wiring code.

The aspect of the experiment relays whose contacts are connected to reactor control must be considered. Reference to the diagram (Fig. 3-75) will show that these relays, except the annunciator, are normally deenergized and "pick up" on trouble. This means that reactor control is lost (except for manual scram) on failure of the experiment's power. The reverse aspect would result in an automatic reactor scram. The connections shown in the diagram are the preferred ones and will usually be safe because of such things as auxiliary power source, cooling medium flow which is independent of experiment's power, or other circumstances. Only in cases where loss of the experiment's power *automatically* creates a

dangerous condition in the experiment will normally energized relays be permitted. (The exception is the annunciator circuits, which are always set up to signal an alarm on loss of either experiment power or reactor-control power to the experiment.)

Experiments which constitute a hazard in case of loss of power may be connected to the 125-kva uninterrupted power source supplied by the MTR.

Experiments which can operate 10 to 20 sec after loss of power without becoming hazardous may be connected to the MTR diesel emergency power supply. This connection is made approximately 13 sec after loss of commercial power.

Thirty seconds after loss of commercial power, the reactor undergoes an automatic scram, and those experiments which can operate without power for this length of time without becoming hazardous will not require emergency power.

Power available from the emergency supplies is necessarily limited, and each experiment is considered individually for connection. Actual connection of experiment control to reactor circuits will be made through a cable and plug. Near each experimental facility around the reactor is a multicontact female socket.

Each experiment will provide cables to this socket. All facilities not used by the experimenter will be left blank.

The purely manipulative aspect of the experiment controls must be considered. The manual power-reduction control must be a spring-return push button. It must be recessed or otherwise protected and labeled to prevent accidental operation. Likewise, switches which control instruments, pumps, a-c power, etc., whose manipulation can lead directly or indirectly to a call for reactor-power reduction, must be treated in a similar manner, rendered inaccessible, or dispensed with. Switches to bypass the individual safety circuits for the purpose of making instrument or operational adjustments will not be used. Reactor release switches or test blocks should be used to accomplish this function where required.

While electrical interference is not yet much of a problem, some cases have occurred. This interference has been of the electromagnetic radiation type, and impulsive noise fed through the a-c power circuits. High-impedance leads should be well shielded and placed in conduit, if feasible.

Instruments susceptible to stray feed from their power circuits should be adequately filtered.

Some general considerations are given here which apply to experimental plugs to be inserted in the reactor tank.

Permissible structural materials for use in the reactor tank or in systems which return effluent to the process-water system are few in number because of corrosion problems. These consist of the following metals and alloys:

Aluminum: Alcoa numbers 2S, 3S, 52S, 61S, and 53S, or in general any aluminum alloy with copper content less than 0.5 per cent.

Stainless steel: AISI types 304, 305, 308, 309, 310, 314, 316, 317, 321, 347.

Zirconium and Zircaloy 2 are also acceptable.

Other materials may be used but must be approved by a qualified metallurgist.

It must be demonstrated that metal surfaces in contact with the process water are below the saturation temperature and that no hazardous situation exists because of high temperatures within the experiment.

Experimental plugs and contents must have an equivalent density greater than water. This would also apply to individual sections of the whole if postirradiation cutting operations are to be made in the canal.

Capsules must be leak-tight and leads for experimenting must be leak-tight at a pressure of 80 psi.

Experimental plugs may be inserted which contain materials having a cross section of 3 cm² per inch of capsule length or less without any further consideration in this regard. For those containing materials of higher cross section, flux depression and values of $\Delta k/k$ must be calculated.

Experimental plugs containing lead tubes or conduits should have such leads at least 30 ft long and should be made of stainless steel or one of the harder aluminum alloys. They should be as small as possible in diameter. The junction between the lead tube and the experimental plug should be able to withstand a considerable amount of flexing. At points of metal-to-metal contact in the tank, neoprene padding should be provided. Outside the reactor tank the lead tube is fitted with a standard head which is connected to pile exhaust. Leads cannot be left open to the atmosphere.

Experimental plugs in the tank must be of dimensions that allow discharge in the reactor-discharge mechanism. This mechanism will accommodate objects $4\frac{3}{4}$ in. in diameter by 14 ft long.

Ordinarily, assignment of an experimental position will permit use, at most, of only the portion of the reactor tank directly above the piece for rigid superstructure.

Mechanisms which cycle fuel or poison into and out of the flux zone should be compensated so that the fuel or poison in the flux zone of the reactor remains constant at all times.

Lines leaving the tank should contain sufficient shielding to reduce the background to instrument tolerance and to prevent attainment of dangerous biological levels under abnormal conditions.

Experimental plugs which present an appreciable contamination or radiation hazard will be monitored for high activity in the exit cooling-water stream.

Some general considerations given to beam-hole experiments follow:

The external dimensions of the plug should be no greater than those of the dummy plugs. The critical length of the plug tip is measured from the stop or shoulder of the plug closest to the tip. Brass slides may be used instead of the rollers which are used in the dummy plugs. Ordinarily it is best to mill keyways on both sides of the plug to give maximum flexibility in scheduling the experiment.

The maximum amount of heat which an experiment may dissipate to the reactor structure is given in Table 3-38. Auxiliary cooling coils have been supplied which will increase this limit, but these are *not* normally activated and *must* be connected if it is necessary to obtain greater cooling. The mechanical design of the aluminum liners is based on a maximum temperature of 570°F in the graphite region. Thermocouples which will permit checking of the important temperatures are available in the reactor structure.

The air flow around the dummy plugs is about 20 to 30 scfm at 100°F maximum.

Calculations of reactivity, flux-depression thermal stress, and thermal expansion must be made.

Shielding should be effective within instrument tolerance under normal operating conditions. All

water-shielded experiments should have reactor setbacks on loss of water flow and should have automatic-valving quick disconnects to facilitate removal of the plug.

Water cooling is available within the cubicle from the MTR experimental facilities lines. This is obtained through flexible lines terminated with Robb quick-disconnect fittings. A process-water or demineralized-water supply may be chosen by valves in the valve cubicle outside the main cubicle. The water may be returned to the process-water system or dumped to the sump. This option is also controlled from the valve cubicle.

Flow and temperature monitors have been placed in the discharge water lines to all experimental facilities. It is only necessary to add additional pressure switches if anything more than the annunciator setpoint is required.

Leads may be taken to and from the cubicle by replacing the doors with temporary shielding or by placing face plates over the holes in the existing shielding doors, feeding the leads through the holes in the plates, and pouring the void space full of lead shot.

Normal utilities are available at the reactor face outside the cubicle.

PART 7 BASIC SAFETY PROCEDURES IN REACTOR OPERATION

INTRODUCTION

3-96 The successful operation of a nuclear-reactor plant depends upon close adherence to all the usual rules of safety plus a number of rules that are unusual for installations not concerned with nuclear reactions and radioactive radiations. It is an interesting fact that, while these unusual aspects of nuclear-reactor operations are potentially far more hazardous than ordinary plant operations, almost all the injuries occurring around such plants are due to nonnuclear causes. The most logical explanation appears to be that at the start of large-scale atomic-energy activities in 1942 it was recognized that it would be necessary to do some vigorous research on biological effects of nuclear radiations and to enforce working regulations strictly to avoid serious radiation damage to personnel. Thus was born, or at least pushed into lively adolescence, the field of activity which was somewhat prematurely labeled "health physics," principally because most of the early work in the development and application of personnel monitoring devices was done by physicists. The very low injury rate due to nuclear radiations thus far in this country is a tribute to the vigor and effectiveness with which the personnel radiation-protection program has been pushed by the Atomic Energy Commission and to the high standards that have been established for the operations conducted in atomic-energy installations.

Operations at the National Reactor Testing Station gave practical recognition to the common objectives of the usual plant safety program and health physics by placing both of these activities under one top supervisor of "health and safety." This man is a chemical engineer by training and has had extensive experience both as a safety engineer and as a "health physicist." The same philosophy regarding the relative functions of the safety engineer and line supervision that has proved so successful in reducing the accident toll in ordinary plant operations works equally well

when applied to the health-physics aspects of nuclear-reactor operation.

The basic safety of nuclear reactors is a matter of great importance for the future development of industrial applications of atomic energy, because both operating costs and distribution of products and services will be more favorable if the reactors can be located in populous areas close to centers of other industrial activity. Because of the lack of complete information on several factors vital to operational safety (and also for reasons of security), it has been necessary in the past to place reactor installations in more or less remote areas. This will probably continue to be the case for some time to come so far as experimental and production reactors are concerned. However, industrial reactor installations must clearly be made safe enough so that they can be located anywhere a commercial feasibility survey might indicate as desirable. Fortunately the progress made during the past several years toward improving the basic safety of nuclear reactors makes this goal attainable.

The operational safety of a nuclear reactor hinges on core geometry, excess reactivity, reliability of control system and coolant, and adequacy of instrumentation. Design engineers are constantly striving to improve all these. All reactors now in operation in this country, as well as those in the advanced development stage, have had a "hazards survey report" prepared, covering the technical aspects of every conceivable situation that might arise affecting the safety of the reactor, its operating personnel, and the surrounding community. These reports are reviewed and discussed in detail with the operating contractor's personnel by a Committee on Reactor Safeguards which acts in an advisory capacity to the Atomic Energy Commission. While such precautions do not constitute absolute assurance that there will never be a serious reactor incident, they do make such an incident less likely by focusing attention on critical problems and encouraging research and development work to solve them.

MTR DESIGN AND OPERATING PHILOSOPHY

3-97 The nuclear characteristics of the MTR required a rigorous adherence in its design to a carefully thought out control philosophy in order to assure reasonably safe operation despite human errors and instrument failures. Basically, this philosophy consists in providing several independent automatic devices for shutting the reactor down and none for starting it. Only the operator can bring the reactor to power, and his actions are at any time rendered ineffective if they do not conform to a predetermined safe procedure. Even if they do so conform, any potential hazard that develops will override the operator's action. Indeed, any control manipulation by the operator which calls for power reduction will override any control-rod movements he may be trying to make to increase the power. All that is lost is time if an error is made.

A detailed description of the MTR control system, embodying this philosophy, is given in Part 1.

SAFETY ASPECTS OF MTR EXPERIMENTAL PROGRAM

3-98 At the MTR the problem of bringing the reactor up to power and holding it there is complicated by the many experiments associated with the operation. From a potential-hazard standpoint these experiments run all the way from none to a level approaching that of the reactor itself. The problem of making the operation of these experiments safe is a very real one and is complicated by the fact that no one knows all the parameters that must be guarded. The only philosophy that can reasonably be applied is to design into the experiment sufficient safety devices, circuits, and procedures to reduce the chances of an incident to a low level commensurate with the hazard involved. This philosophy is implemented by having each experiment require the prior approval of a program committee representing all departments involved and a reactor safeguard committee representing all technical sciences involved and by utilizing the services of a project-engineering group to engineer the proper safeguards into the experiment and an operations

group to see that the experiment is installed and operated within the prescribed framework.

A high degree of protective instrumentation may well generate more trouble than the machinery being protected. This is a real problem at the MTR because of the large number of experiments and is made more acute by virtue of the fact that in reactors operating at high-power density a restart may be impossible after even a short shutdown because of the rapid growth of fission-product poisons when power is reduced. To assure even a reasonable continuity of operation, it is necessary to be able to free the reactor from false calls for power reduction when it has been ascertained that the apparent trouble is due to instrument failure and therefore not real. This is accomplished by a series of instrument "test blocks" which remove the offending instrument from its connection with reactor control. This can be, and is, a safe procedure only if:

1. Operators are trained to remove test blocks only when they are sure the trouble is not real but due to instrument malfunction.

2. Only those parameters whose time constant is sufficiently long to permit manual monitoring by an operator in continuous attendance are provided with an override switch or a test block. Cases where it is not reasonable to assume that an operator has ample time to take appropriate action before real danger develops are arranged so that their safety devices cannot be violated.

3. Removal of any safety circuit from reactor control alarms an annunciator at the control console which cannot be cleared until the safety circuit is restored.

One final device which contributes both to the safety and continuity of operation is the use of an extensive annunciator system. Two aspects of this system should be emphasized. (1) The alarm should sound before conditions have reached the point where drastic correction is required, but not for mild departures from normal which are self-correcting. Otherwise repetitive alarms will generate the condition where operators ignore the alarm. (2) Alarms of any nature should be cleared as soon as possible. The operators should never be allowed to get accustomed to operating with "red lights on the board," because the permitted loss of a few safeties increases the operational hazards unnecessarily.

The operation of a nuclear reactor is no more hazardous than that of any other industrial plant if the control and safety devices are carefully thought out and rigorously maintained even though they may make continuous operation a little difficult at times. That the problems can be solved is attested by the smooth and essentially continuous operation of the MTR over the past two years.

In addition to the safety aspects of reactor operation which are directly concerned with controlling a chain reaction, there are other factors which are unique in the atomic-energy business.

One of these is the problem of handling the extremely radioactive material after it is irradiated in the reactor. Normally the materials which give off the most intense radiation are the irradiated fuel units. At the MTR it is estimated that these would read 40 million r/hr at 1 ft in air, although no one has been able to make a direct measurement. This radiation level would give a lethal dose to humans in a small fraction of a second; so special equipment and procedures must be used to prevent overexposure of personnel. At the MTR, as in other reactors, the fuel is discharged by means of specially engineered mechanisms which allow the operators to do all their handling without being exposed to the radiation. The operator works with the fuel with long tools through 18 ft of water in the reactor tank and lowers the fuel units into a canal under the reactor. The canal is simply a deep concrete trench filled with water. Most reactors that operate at an appreciable power level are provided with such a facility as a storage area for irradiated material. The water allows the operator direct observation of the irradiated material and still provides adequate shielding.

When the fuel elements have decayed to some extent in the canal, it is necessary to ship them to a chemical processing plant for recovery of the remaining uranium content. This is accomplished by lowering a very heavy lead-filled carrier into the canal, filling it with fuel elements, replacing the lid, and removing it from the canal. At the MTR this carrier weighs approximately 11 tons and provides 14 in. of lead shielding. The carrier is then transported to the chemical plant, where it is lowered into another canal and the fuel elements are removed and stored until they are

processed. This same handling technique is used for any irradiated material that must be shipped to other locations. The entire procedure is quite simple, and there have been no instances to date of overexposure or other difficulties.

The handling and storage of fuel elements prior to and after utilization in the nuclear reactor presents a special safety problem that is common to all reactors employing enriched uranium. Such fuel must of course always be transported and stored in subcritical units. This is achieved both by suitable space separation and the use of neutron-absorbing liners and spacers in the storage racks.

MAINTENANCE PROCEDURES

3-99 The maintenance procedures around a nuclear reactor must take into consideration the radioactive radiations associated with the operations. These radiations persist even after the reactor has been shut down, in the case of those materials or equipment parts that have been exposed to neutrons. In so far as possible the reactor is designed so that those pieces of equipment likely to require frequent maintenance are kept out of the neutron atmosphere. In general, the maintenance problems in nuclear-reactor plants have thus far been fewer and less difficult than those associated with the operation of chemical plants for the recovery of fissionable material from irradiated nuclear fuel. This may not be the case, however, for the homogeneous solution type of reactor or for nuclear-power reactors employing liquid metal coolants.

The "safe-work permit," which is mandatory at the MTR before maintenance on the reactor or any of its components can be undertaken, assures that the level of radiation to which maintenance personnel are exposed is within permissible limits for the time required to do the assigned job. Any equipment that has become inoperable may be repaired in a "warm shop" if it is only slightly radioactive. If it is highly radioactive, it may still be possible to effect the necessary repairs by remote-control means inside a "hot cell." If not, a replacement part must be provided. Disposition of the defective equipment may take one of two possible routes. If the equipment is sufficiently valuable and the radioactivity sufficiently

short-lived, it may be stored in a shielded location until the activity dies down to the point where repair work is possible without undue exposure to personnel. The alternative is to send it to the burial ground.

HEALTH-PHYSICS ASPECTS OF REACTOR OPERATION

3-100 Since a nuclear reactor operating in upper-megawatt power levels generates an amount of radioactivity tens of thousands of times greater than all the radium refined in the world prior to World War II, it is clear that the protection of operating personnel and surrounding populations from effects due to the radiations, or to secondary types of radiation, resulting from it is the number-one problem of any reactor plant. In the absence of a reactor incident which might release large quantities of radioactive fission products, the principal factors requiring attention are the following: (1) protection of operating personnel from direct radiation from the reactor, which is accomplished by bulk shielding; (2) protection of operating personnel and surrounding populations from excessive radioactive products in any reactor coolants, either air or water, leaving the plant; (3) protection of operating personnel during the handling of radioactive material after exposure in the reactor. Because of the large volume of experimental work at the MTR and the very high neutron flux involved, there is also (4) the necessity for continuous surveillance of experimental equipment for leakage of radioactive products into the reactor-building atmosphere or reactor coolant streams.

The basic elements of the personnel radiation-protection program are (1) personnel metering, (2) area monitoring, (3) limitations on permissible amounts of radiation exposure, and (4) prevention of inhalation or ingestion of radioactive materials.

Personnel metering devices are carried upon the person of the individual to be metered and, although they give no warning of excessive exposure, are able to record the amount of radiation received. The most important of these devices is the film badge, which is usually a small package of two types of dental x-ray film, one being considerably more sensitive to radiation than the other. The film can be calibrated for radiation-

exposure density and thus gives a measure of the amount of radiation received by the individual carrying it. In most plants, film badges are worn for one week at a time, after which the film is changed, developed, and read and the results are permanently recorded. Pocket chambers are condenser-type devices which are worn routinely by all personnel in areas of possible radiation hazard and are read on a separate instrument each day. Although less accurate than the film badge, they give a daily indication of probable exposures. Several other personnel metering devices are used as occasion warrants: (1) film rings for measuring exposure to hands and forearms; (2) direct-reading dosimeters, which are quartz-fiber electroscopes so designed as to enable the individual wearing them to read his total exposure at any time; (3) fast-neutron film badges, which are special adaptations of the ordinary film badge to make them sensitive to fast neutrons; (4) slow-neutron chambers, similar to the ordinary pocket ionization chambers but boron-lined to make them sensitive to thermal neutrons.

All the information obtained from film badges, pocket chambers, and special devices is generally recorded on permanent cards so that the total exposure of a given individual for any desired period of time can be readily obtained and his average exposure thereby controlled within safe working limits set by various professional committees and government agencies, based on extensive research programs.

For special types of atomic-energy work, protective clothing of one type or another is provided. Coveralls, gloves of several kinds, caps, shoes, shoe covers, boots, etc., are available for use where needed. Careful control of such clothing to assure against the spread of radioactive contamination is of major importance. To make such clothing effective and to protect personnel from external contamination, roped-off areas, step-off mats, and similar devices are used.

The use of respirators in areas where air-borne contamination is present is essential to the prevention of internal contamination. In most instances a simple filter-type respirator is adequate. In cases of severe air contamination, however, air or oxygen must be used.

In areas where it is planned to carry out work of a rather unusual nature or entailing unusual

equipment or procedures, radiation surveys of the equipment and area are carefully made prior to the beginning of actual work. In addition, monitoring during the work is often required. A work-permit system is most effective in making certain that adequate presurvey and monitoring are carried out. Under this system the notification of health-physics personnel, their prejob survey work, and their monitoring service are required before final approval to begin such a job can be obtained. Overexposure of personnel to direct radiation can be prevented by any one or a combination of time limitations, distance from radioactive source, and shielding.

ROUTINE SURVEYS AND MONITORING OF WORK AREAS

3-101 In order that personnel can be kept informed of what areas, materials, and equipment can be radiologically hazardous, it is necessary that radiological surveys be made on a routine basis. In most reactor and laboratory areas, routine surveys are made at rather frequent intervals. Two different techniques are used: (1) a portable instrument is used for the purpose of determining the levels of direct radiation prevalent in normal working areas; (2) a method called smearing, which is a procedure of wiping with small circular papers and counting the contamination collected thereon, is for the purpose of detecting excessive amounts of contamination on floors, desks, working surfaces, etc. A combination of these two procedures can result in careful records of possible radiological hazards.

In order to make certain that radioactive and contaminated materials and equipment are not haphazardly shuffled among various areas, certain requirements regarding transfers and shipments of all items from "hot" to "cold" areas are generally required. These procedures are kept as simple as possible and yet usually require checking and tagging by qualified health-physics personnel. Authorizations of repairs to equipment possibly radiologically hazardous require similar health-physics attention.

Personnel can, however, unknowingly transfer contamination from one area to another. Consequently, several devices are in general use for the purpose of checking individuals as they move

from one area to another. The most common such device is called a hand and foot counter. With the cooperation of personnel this instrument will indicate whether or not a particular individual is carrying contamination on his hands or feet or both. Unfortunately, beta-gamma emitters and alpha emitters cannot be checked on the same equipment. Both types of instruments must be provided in areas where both types of radioactive materials are present. Another kind of checking device for personnel is called a *frisker*. This is a series of Geiger tubes so placed as to guard a particular doorway or passageway. One or more of the tubes will pick up radiation from contamination on a passing individual. This will release an electrical impulse to a central device, which in turn will sound an alarm if the contamination is excessive. Several specialized devices, all based on the frisker principle, are in use. A particular one called a *styletron* actually blocks the doorway and cannot be released until the contaminated individual has released it by entering a "clean-up" area.

It is often necessary to know of any changes in radiation levels that might occur over a long period of time or during accomplishment of a particular job. In areas regularly frequented by personnel, instruments are used for routine and constant monitoring. One of the most useful for direct radiation is the *monitron*, an instrument which both indicates and records the radiation received by it. The monitrons are generally equipped with alarm devices which operate when the radiation level exceeds a certain preset amount.

Air contamination can be an even more serious hazard than a comparable direct radiation field. In order to detect levels of air contamination that might be hazardous, a device called a *constant air monitor* or *mobile air monitor* is in general use. Major components of such a device are an air pump, a filter surrounding a Geiger tube, a count-rate meter to which the tube is attached, and recording and alarm equipment. The rate at which radioactive particulate matter builds up on the filter is utilized for measuring and recording the quantity of particulate radioactive material in a given volume of air.

Special monitoring procedures are often used for work in areas not routinely entered or on

equipment not worked upon. Several types of portable indicating devices, usually rate-type instruments, are used for this purpose. The most common ones are beta- and gamma-sensitive only and include ion chambers, Geiger-Müller instruments, and scintillation detectors. Where alpha contamination is possible, a very thin windowed ionization chamber or proportional counter serves the purpose best. For the detection of neutron fields, several instruments have been developed, none of which is wholly satisfactory. All depend upon some type of proton recoil or nuclear absorption-energy emission reaction, and inasmuch as neutron energies are as important as the total number of neutrons in the field, it becomes very difficult to measure the potential physiological damage. Such potential damage is, however, generally estimated by more than one procedure at any particular time so that an overall estimate of fair accuracy is obtained.

Air contamination in areas and at times not routinely monitored by constant operating devices is usually checked by taking spot samples. Three general methods of obtaining these are in use:

1. Filter devices by which air is pumped through filter paper which is subsequently counted.
2. The "precipitron," in which a high static charge attracts particulate activity to one electrode in an electrostatic system through which air is flowing.

3. Samples collected by the impingement technique.

A particularly effective device for locating sources of particulate contamination in air is an attachment for the constant air monitor called a *snooper*. This very simple attachment is in reality just a hose which can be attached to the air intake on a constant air monitor and by manipulation can be made to indicate the exact source of particulate air contamination.

CONCLUSION

3-102 Several of the more obvious safety aspects of reactor operation have been discussed. To be sure, they are very real hazards, but except in a few isolated instances they remain as potential hazards only, since the original design of the installation and the operating procedures adequately protect the equipment and personnel from harm. As in the case of any other activity where there is a great potential hazard calling for special rules and regulations, it is the little things that jeopardize a safety engineer's peace of mind and the plant safety record. Improper use of hand tools, poor lifting techniques, and the many other little acts of carelessness resulting in cuts, bruises, and strained backs comprise the same major problem around nuclear-reactor plants as at other installations.

APPENDIX A

MTR HEALTH-PHYSICS PROCEDURES

EDUCATION IN HEALTH PHYSICS

3-103 A major problem is the education of all MTR personnel in the general principles of health physics and the dissemination to both resident and visiting personnel of complete information with regard to health-protection regulations applicable to MTR plant operations.

A 3-hr health-physics indoctrination lecture is presented to each new employee, which explains the basic health-physics policies and regulations in order to promote radiation safety on the individual level.

A generalized information booklet on the health-physics program at the MTR is distributed to each new employee when he reports for work. It serves as the health-physics guide for all employees.

Direct responsibility for the dissemination of adequate health-physics information resides in the health-physics branch.

Responsibility for seeing that employees are adequately trained in health physics for the job to which they are assigned, and for assuring that operational and experimental work is conducted in accordance with the best known safety practices, resides at all times with the supervisors in charge of the operations involved.

AREA SURVEY AND MONITORING

3-104 Routine Daily Surveys. Daily surveys with a GM survey instrument or a "cutie-pie" type ionization chamber, or both, are made for all laboratories and other areas in which radioactive materials are handled routinely. Such daily surveys are usually accomplished by the

health physicist on the evening or night shift in order to interfere as little as possible with normal operations. Any surface or article reading in excess of the maximum permissible levels is appropriately tagged or labeled.

3-105 Special Surveys. Any changes or additions in the experimental facilities at the reactor itself are monitored carefully by the health physicist on duty. At the completion of such changes, the area of the new facilities is surveyed and a permanent record of the new levels of all observed types of radiation is made and attached to the record of the last routine survey. As time permits, a survey for any and all types of radiation is made in any area upon the request of the supervisor in charge. Normally such results are entered only in the log, unless they differ appreciably from past results, in which case a permanent record is appended to the results of the last routine survey of the area.

Upon notification of intent to work in a known or suspected radiation field, the health physicist on duty surveys the area and monitors the operation to an extent sufficient to determine a working time which will keep each individual's exposure under the daily maximum permissible exposure. In case of emergency, and following conference with the supervisor in charge of the operation, weekly tolerances may be permitted. A suitable record of the exposure field and working time is made and given each individual so exposed. Such information also is entered in the health-physics log.

When issuing work orders, supervisors initiate safe-work-permit forms, if the potential radiation hazards of the work indicate that health-physics

service will be advisable. The safe-work permit is brought to the health-physics office where the health-physics monitor fills in the appropriate spaces indicating to the worker the special clothing to be worn, special personnel monitoring instruments needed, exposures to be expected, time to spend on the job, what respiratory protection will be used, the health-physics services that will be required, and any special instructions that are appropriate. The operations branch retains a copy of the safe-work permit at the operations-control desk to prevent changes in operational status of the reactor which will create new radiation hazards while the work is in progress. The safe-work permit is terminated by operations when the work is finished.

3-106 Smear Samples. All areas in which surface contamination is possible are subjected to routine smear sampling and counting.

Schedules for routine smear sampling vary in the various areas at the MTR, depending upon the probability of radioactive contamination and the advisability of keeping any particular area free of radioactive contamination. Generally, smear sampling is done weekly in the first-aid room, all eating areas, and "hot" areas. Other areas are sampled less frequently, but on a routine basis.

PERSONNEL MONITORING AND PROTECTION

3-107 Personnel Monitoring Devices. Three major types of devices are used for personnel monitoring: film badges (with beta-gamma sensitive film), pocket meters, and direct-reading dosimeters. Film badges and pocket meters are issued to and worn by all persons admitted to the MTR area. In addition to beta-gamma film badges, separate badges containing neutron-sensitive film are supplied to all personnel working regularly in areas where neutron exposure is anticipated. Those not regularly employed in such designated areas wear neutron film badges when required to work in such an area and in any area at the request of the health physicist. Neutron film badges are issued as needed from the health-physics office and deposited there at the end of the shift.

Both the film badge and the pocket meters are worn on outer clothing and as close together as practicable. A reasonable amount of care should be taken to prevent damage and to eliminate accidental discharge of the pocket meters. Shielding of the meters should be avoided as much as possible.

Film rings are issued by the health-physics branch for use in the MTR area. Such issue is made at the request of the individual concerned or at the discretion of the health physicist on duty.

Direct-reading beta-gamma dosimeters are issued to individuals by health physics on request or at the health physicist's discretion. The dosimeters generally prove valuable in limiting accumulated exposures when personnel are working in variable radiation fields.

3-108 Pickup and Reading Schedule. Film from the film badges is replaced every week end. The exposed film is developed and compared to calibration film to determine the exposure that each person has accumulated during the past week.

If at any time during a calendar week the total pocket-meter readings for any one man exceed 300 mr, or if the pocket meter indicates more than 100 mr in 1 day, the corresponding film badge is immediately developed. The health physicist on duty makes certain that a new pack of film is inserted and the badge is ready for wearing before the individual next reports for duty.

Every case of exposure in excess of 300 mr per calendar week, as indicated by film results, is investigated by the health-physics branch and a permanent record made of the circumstances.

Smear samples are made with small circular filter papers. Approximately 5 lb pressure is applied, and an area roughly 1 ft² is covered. The health physicist smears those areas which in his judgment are most likely to have been contaminated, one smear sample being taken for approximately each 25 ft². The usual permanent items of equipment in each area are smear sampled. This includes floors, desk tops, table and hood working surfaces, and similar items. A map-type record of smears is prepared at the time of sampling. Portable equipment, such as instruments,

glassware, etc., is smear sampled upon request, before transfer to another locality, or before repair or alteration.

3-109 Permanent Monitor Installations.

Permanent monitoring devices are installed at appropriate locations throughout the area. Such devices include hand and foot counters, constant air monitors, monitrons, and GM friskers.

All hand and foot counters are serviced daily by changing the protective paper and checking their operation; they are recalibrated, by means of a standard source, weekly.

Constant air monitors also are serviced daily by changing the filter paper and recording the change on the chart. They are checked, for proper operation and for any indication of high radiation level, during each 8-hr period by the health physicist on shift. Calibration is accomplished once each week by means of a known source.

Monitrons and friskers are rough checked daily by exposure to a source sufficiently high to cause their normal reaction and alarm.

3-110 Anticontamination Clothing. Anticontamination clothing is provided all persons in the exclusion area who work in the immediate reactor vicinity or enter any of the laboratories designated as "hot." Shoe covers are provided all regular employees of these areas; shoe covers are worn by all visitors to "hot" labs. The health-physics office, the process-water building, the fan house, and all other structures in the exclusion area except other offices, counting rooms, cold shop, and "cold" labs in the reactor-building wing have the same requirements as the reactor area. Anticontamination clothing is not allowed in the other offices, counting rooms, cold shop, and cold labs excluded in the previous sentence. Anticontamination clothing is not worn outside the exclusion area.

In general, all laboratory personnel are issued laboratory coats. Operators, maintenance men, and similar personnel wear coveralls. All those regularly issued either type clothing are also provided with shoe covers. Laboratory coats and cotton shoe covers are provided all visitors to hot labs or to other sites when there is any cause to suspect contamination.

Cotton or rubber gloves are issued to all persons handling radioactive materials.

Expendable head coverings are issued and worn at the discretion of each individual supervisor.

Personal clothing contaminated under circumstances where expendable clothing should have been worn, and would have prevented the contamination, is not replaced.

If at any time an employee suspects that he, or any item of his clothing, might have become radioactively contaminated, he immediately reports to the health-physics office for surveying. Any radioactivity in excess of the maximums listed in the following appendix will result in decontamination or confiscation, or both, of the contaminated article or articles of clothing.

3-111 Masks and Hoods. Respiratory protection is available for all emergencies, to be used by all personnel exposed to possible radioactive atmospheric contamination. Three types are provided, the respirator (filter) type, air-supplied masks, and self-contained air or oxygen units.

3-112 Contamination Procedure. Any person who has been engaged in an activity which might result in contamination of himself or his clothing must make certain that his person is adequately surveyed as he leaves the job. Such a survey may include the use of a hand and foot counter, use of detection devices on loan from the health-physics branch, or a visit to the health-physics office, or all three. Considerable discretion must be used by the individual, and he is encouraged to avail himself of the services of a member of the health-physics branch whenever he suspects contamination.

3-113 Prohibitions. Eating or drinking is prohibited in the exclusion area, except at times and in areas specifically approved by the project manager and the health-physics supervisor. "No Smoking" signs are posted by the health-physics staff in areas where it is deemed necessary in the interest of personal health and safety. Smoking is prohibited within the area designated or, if not specifically designated, within 25 ft of such a sign. It is strongly recommended that hands be washed before eating or smoking and before leaving the exclusion area. No returnable containers for

bottled drinks, either full or empty, may be taken into the exclusion area.

Any person with an open wound or sore is prohibited from working with radioactive materials except as specifically authorized by his supervisor and the health physicist on duty. In such case every precaution is taken to prevent any contamination of that body area. In the event that injury and contamination occur simultaneously, decontamination procedures are instigated immediately and medical aid is obtained as soon as possible.

3-114 Urinalysis. Urinalyses are run at regular intervals. In the event of an emergency, special samples may be taken for immediate analyses.

INSTRUMENTS

3-115 The following general types of instruments are used by the health-physics branch and other qualified personnel requiring radiation-detection instruments:

1. Cutie-pie type ionization chambers to be used for routine high-level beta-gamma survey.
2. GM-type survey meters for routine low-level beta-gamma survey.
3. Film badges, pocket meters, dosimeters, film rings, and proteximeters, as discussed in Secs. 3-107 to 3-114, for personnel monitoring.
4. Constant air monitors for detecting appreciable increases in air-borne particulate radioactivity.
5. Monitron beta, gamma, and neutron monitoring devices for detecting any general increase in the radiation level.
6. Hand and foot counters for surveying hands and shoes of individuals for beta, gamma activity.
7. Alpha "poppies" for the detection of alpha activity, particularly on personnel.
8. A beta, gamma counter scaler for counting beta, gamma smear samples.
9. Fast-neutron dose-rate meters for surveying and monitoring of fast-neutron radiation.
10. "Juno" type alpha, beta, gamma survey meters are used primarily for surface alpha surveying.
11. "Fish-pole" probes with beta, gamma ionization chamber-type survey instrument.

12. A small disk-type air monitor for rapid sampling of air-borne particulate matter.

13. Boron trifluoride neutron scanner for surveying thermal-neutron radiation.

All these instruments are used by the health-physics branch or others as needs arise for their particular abilities. Personnel monitoring devices are, of course, used primarily by the individuals to be monitored. "Cutie pies" and GM beta-gamma survey meters may be placed on loan from the health-physics branch to any MTR supervisor whose section has a continuing need for them.

Maintenance of all truly portable instruments is accomplished by central facilities. Such instruments are calibrated, also by central facilities, at least once a month. More nearly permanent devices, including monitrons, constant air monitors, hand and foot counters, and disk-type air monitors, are calibrated weekly by the health-physics branch and repaired by the MTR instrument development group.

Locations of permanent monitoring devices are determined by the results of routine area surveys.

DUTIES OF THE SHIFT HEALTH PHYSICIST

3-116 The primary reason for the assignment of a health physicist to each shift is to make certain that health-physics advice and experience are available in the event of an emergency at any time. In such a crisis all other assigned duties are of secondary importance, and full time and attention are devoted to the emergency until operations become normal or the health physicist is relieved of duty.

When any emergency involving radiation exposure or contamination occurs, the following are the principal duties of the health-physics shift representative:

1. If physical injury has occurred to any person, and medical personnel are not present, take charge of all first aid and transportation of the injured.
2. With appropriate instruments, locate the exposed or contaminated area immediately and take all possible steps to prevent the spread of contamination.
3. Mark, tag, or otherwise identify all areas unsafe for extended work. Notify each supervisor who is currently in charge of such an area.

4. Survey each individual exposed to contamination and make certain that all contaminated clothing is removed and that his person is clean.

5. Notify directly the health-physics supervisor or the chief health physicist, or both, of the emergency. Have the following facts ready: (a) time, (b) place, (c) personnel involved, (d) approximate type and degree of contamination, (e) steps taken to meet the emergency and progress toward cleanup.

6. If neither the health-physics supervisor or the chief health physicist can be contacted and the emergency is extreme, then make certain that the project manager and the supervisor of the emergency area are notified.

7. Continue monitoring and surveying until the cleanup process has been completed. Survey all decontaminated equipment and all items used in the emergency area.

8. Recommend decontamination procedures and be prepared to state the external and internal dosages received by all personnel working in the emergency area.

9. Prepare a written report of the emergency. This report should include all the information accumulated in following the foregoing directions. It should specifically state times, places, names, items, and radiation levels and should also contain a descriptive summary of the accident itself.

Routine duties of the shift health physicist include those normal to the usual responsibilities of the health-physics branch. Every area which might be considered hazardous from the radiation standpoint is visited at least once on each shift. Areas in which radioactive materials are normally handled, and in which day-to-day variation of sources is to be expected, are also surveyed with appropriate instruments. The shift health physicist makes himself available to all personnel on duty for consultation on radiation problems and surveys or monitors as requested by any supervisor and as time permits. He may survey and recommend on his own initiative, or at the request of any employee, if he believes the work to be justified. He makes certain that all his activities have been entered in the health-physics log before he departs at the end of his shift.

WASTE DISPOSAL

3-117 Dry Wastes. All dry radioactive waste is accumulated at the MTR site in special "hot" waste containers. The waste containers are disposable and, twice weekly, the hot waste is picked up and disposed of at the burial ground by a crew from general maintenance. Special pickups of hot waste may be obtained, on request.

3-118 Contaminated Equipment. Larger items of equipment or material which are suspected of radioactivity too great to permit their use are surveyed by a member of the health-physics branch, who tags each item appropriately with his name, the date, time, and radiation level. Such items are disposed of by burial or storage in a radioactive salvage location.

3-119 Liquid Wastes. Liquid radioactive wastes are disposed of by any of several methods, depending upon the quantity, concentration, type of radiation, and half-life. They may be emptied into the effluent pond, the retention basin, the holdup tanks, or special shielded containers to be delivered to the chemical processing plant. These alternatives have been listed in order of increasing radioactivity of wastes.

Responsibility for advice regarding the disposal of any radioactive waste rests with the health-physics branch. Actual disposition, however, will be arranged, in so far as possible, by the original custodian of the material.

Routine records are maintained of the activity of the effluent water and the stack gas. The activity of the effluent water is kept below the maximum permissible concentration of gross activity in drinking water after 188 days' decay (188 days is the estimated minimum flow time of underground water from the MTR to the nearest unmonitored well). The activity of the stack gas is kept below the concentration which, after atmospheric dilution, could cause the maximum permissible exposure to any individual on or off the site.

DECONTAMINATION

3-120 Decontamination procedures vary with several aspects of the individual problem. Three

general classifications can be established: personnel, equipment, and buildings and areas. Each needs a different approach, which may still vary with the type of radioactivity involved.

Whether instruments, tools, equipment, etc., which have been contaminated shall be cleaned, salvaged, or disposed of by burying depends upon the value of the material, the type of radioactivity present, and the ease or difficulty of decontamination. If the contamination is external only, soap and water, acid, or wet abrasion, or all three, may be used for its removal. Various combinations of these may be used in accordance with the materials of construction. It may sometimes be necessary to remove and replace a portion of the equipment, disposing of the removed material by burying. Small items of equipment may be decontaminated by their regular custodian in any hot sink; larger items, however, are decontaminated by Maintenance in a special area set aside for this purpose. Persons engaged in such work must take adequate precautions and wear suitable protection (such as expendable clothing, boots, gloves, respirators, etc.) to prevent their own contamination, either external or internal.

Buildings which have become contaminated are subjected to decontamination procedures similar to those suggested for equipment, except, of course, on an appropriately larger scale. The maintenance branch is normally responsible for such work, and the maintenance supervisor sees that his personnel are adequately protected during the job. The protection of each person in the area is the responsibility of his individual supervisor. The health-physics branch is responsible for surveying, monitoring, establishing working times, and advising decontamination procedures in such instances.

Contaminated outside areas normally are decontaminated by removal of sufficient material to include all, or nearly all, the radioactivity present. This may entail the removal of large quantities of earth or concrete. If such an excavation reaches sufficient depth that refilling gives adequate shielding of the remaining radioactivity, and the half-life of the material is within the range of a few years, complete removal may not be necessary. Each such instance is handled as a special case and the final decision reached by consultation with appropriate authorities.

Cleanup responsibility for all spills normally rests with the supervisor of the section in which the spill occurs. This means that the person causing or allowing the spill takes immediate steps to clean up and decontaminate. Health Physics is notified at once and assumes responsibility for survey, monitoring, warning of all other personnel, and advising regarding decontamination. If the spill is large and of a serious nature, the responsible supervisor may request assistance from Maintenance. The responsibility, however, cannot be passed.

TRANSFER OF MATERIALS

3-121 No instrument, tool, or item of equipment, once having been used inside the exclusion area, is removed therefrom without first having been surveyed and appropriately tagged by a member of the health-physics staff. On the tag is recorded the date, name of surveyor, maximum radiation level measured, and any pertinent remarks concerning removal of the item. Recommendations on handling are normally also given verbally to the person concerned. Such recommendations are in accordance with AEC regulations for transfer to other points of the reactor testing station and ICC regulations if sent to any point outside this area. The responsibility for proper handling rests with the supervisor in charge of the material to be removed.

Shipment of test samples which have been irradiated or other radioactive materials are in general handled as described above.

Additional procedures are in effect which assure that all radioactive shipments meet the detailed regulations specified by the Interstate Commerce Commission, the Bureau of Mines, and the Atomic Energy Commission.

A permanent health-physics radioactive-shipment survey record will be maintained on all radioactive shipments leaving the MTR exclusion area.

RECORDS AND REPORTS

3-122 General Health-physics Activities. The health-physics branch maintains such records as are necessary to keep a complete summary of its activities, to indicate the radiation

exposures received by all personnel within the MTR exclusion area, to record the radioactive condition of all items leaving the MTR site or transferred therefrom, to summarize the radiation level of all area and buildings on the site, and to have available the complete history of every major radioactive spill or exposure occurring at the MTR site.

A running log is kept in a bound record book of all special and routine work done by any member of the health-physics branch. The entries consist of the date, clock time, name of surveyor, name of person requesting the work, persons or items surveyed, results, recommendations made, and any other information or observations that are deemed of possible importance. The entries are made as soon as possible after a job is completed and in any case before the health physicist completes his shift. Possible exceptions might be envisioned where the results of smears are not immediately available, or in cases of emergency when a job is not completed as the shift changes. In such cases sufficient space is to be left in the logbook to make the entry when the information is complete. Entries are as nearly in chronological order as is feasible. Logbooks when filled become part of the permanent records of the health-physics branch.

3-123 Personnel Records. Radiation exposure records are kept on all persons on whom such records are available, i.e., those who are wearing or have worn film badges and pocket meters. An effort is made also to obtain exposure records accumulated for any employee prior to his employment by Phillips Petroleum Company. A total lifetime exposure record is to be kept for each individual, and therefore a knowledge of his preemployment exposure is desired.

3-124 Equipment and Shipments. Radiation records on each shipment leaving the reactor testing station site are kept by the shipping clerk. Records on material transferred within the reactor testing station area are kept only if deemed advisable by the health-physics supervisor. An example where such a record might be desired would be equipment consigned to burial grounds if salvage at a later date is contemplated.

Records are kept on all instruments under the

jurisdiction of the health-physics branch, whether on temporary loan from central facilities or permanently a part of the MTR facilities. Such records give the currently assigned location of the instrument, last calibration and servicing date, and any unusual behavior or circumstance pertaining to the instrument.

3-125 Operational Records. Permanent records are kept by the Health-Physics Branch of stack-gas activity, process-water activity, monitor readings, and the activity of liquid waste taken from the several sampling points of the liquid-waste disposal system.

APPENDIX MAXIMUM PERMISSIBLE RADIATION LEVELS AT THE MTR

Exposure of Personnel to External Radiation

Type of radiation	Permissible exposure
Gamma....	Whole body—300 mr/week (60 mr/day) Skin—600 mr/week (120 mr/day) Hands and forearms only—1.5 r/week
Beta.....	Whole body—300 mrep/week (60 mrep/day) Skin—600 mrep/week (120 mrep/day) Hands and forearms only—1.5 rep/week
Thermal neutron.	Whole body—1750 n/cm ² /sec for an 8-hr day
Fast neutron....	Whole body—50 n/cm ² /sec for an 8-hr day

Note: The fast-neutron dose-rate meter should be well integrated with the neutron-energy dose-rate curve. It will probably be calibrated in rem and the permissible level for all neutrons will then become 300 mrem/week (60 mrem/day).

Beta-Gamma and Alpha Contamination

In air:

For breathing without masks

CLL *..... $1.5 \times 10^{-11} \mu\text{c}/\text{cm}^3 \uparrow$
First count..... $3.0 \times 10^{-9} \mu\text{c}/\text{cm}^3 \uparrow$

In water:

Into effluent pond—
after 188 days' decay.....

Must not exceed $10^{-7} \mu\text{c}/\text{ml}$ for α, β, γ

* CLL refers to long-lived component.

† Eight-hour maximum permissible level (multiply by 3 to obtain 24-hr maximum permissible level).

	Survey meter β, γ	Smear, * c/m	
		β, γ	α
Hand count (palm or back).....	400 c/m		
Personal or expendable clothing.....	500 c/m		
Personal shoes, outside.....	1000 c/m		
Hot lab (tables, floors, etc.). Post areas.....	>7.5 mr/hr	<200	Background <10
Material to shops.....	<1 mr/hr	<50	<10
Equipment leaving MTR.....	<0.5 mr/hr	<50	<10
Commercial carriers leaving MTR ICC or Air Regulations			
AEC controlled carriers.....	Safe handling	<200	<10
Salvage materials.....	<0.2 mr/hr	<20	<10
Material to warehouse, etc.....	None detectable	<20	<10
Offices and canteen.....	None detectable	<20	<10
Floors in hallways, storage rooms, cold labs, table tops, janitors' closets, furniture.....	<0.2 mr/hr	<50	<10

Instruments to calibration will have no loose contamination; if fixed contamination is present, the instrument will be wrapped and appropriately labeled.

* Beta smears counted at approximately 10 per cent geometry, alpha smears counted at approximately 50 per cent geometry.

APPENDIX B

DRAWINGS

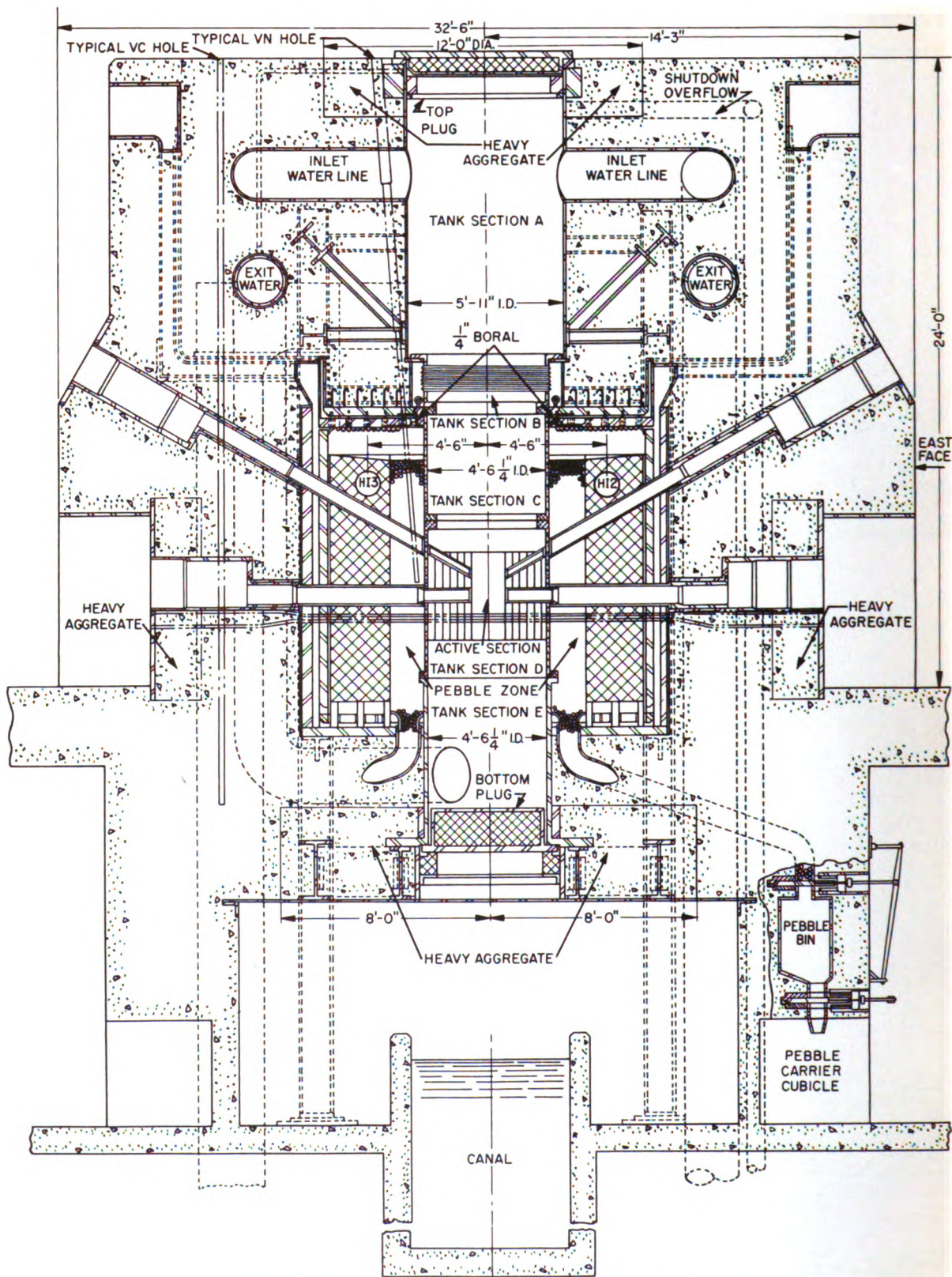


Fig. 3-78 Vertical section through east-west center line of reactor structure.

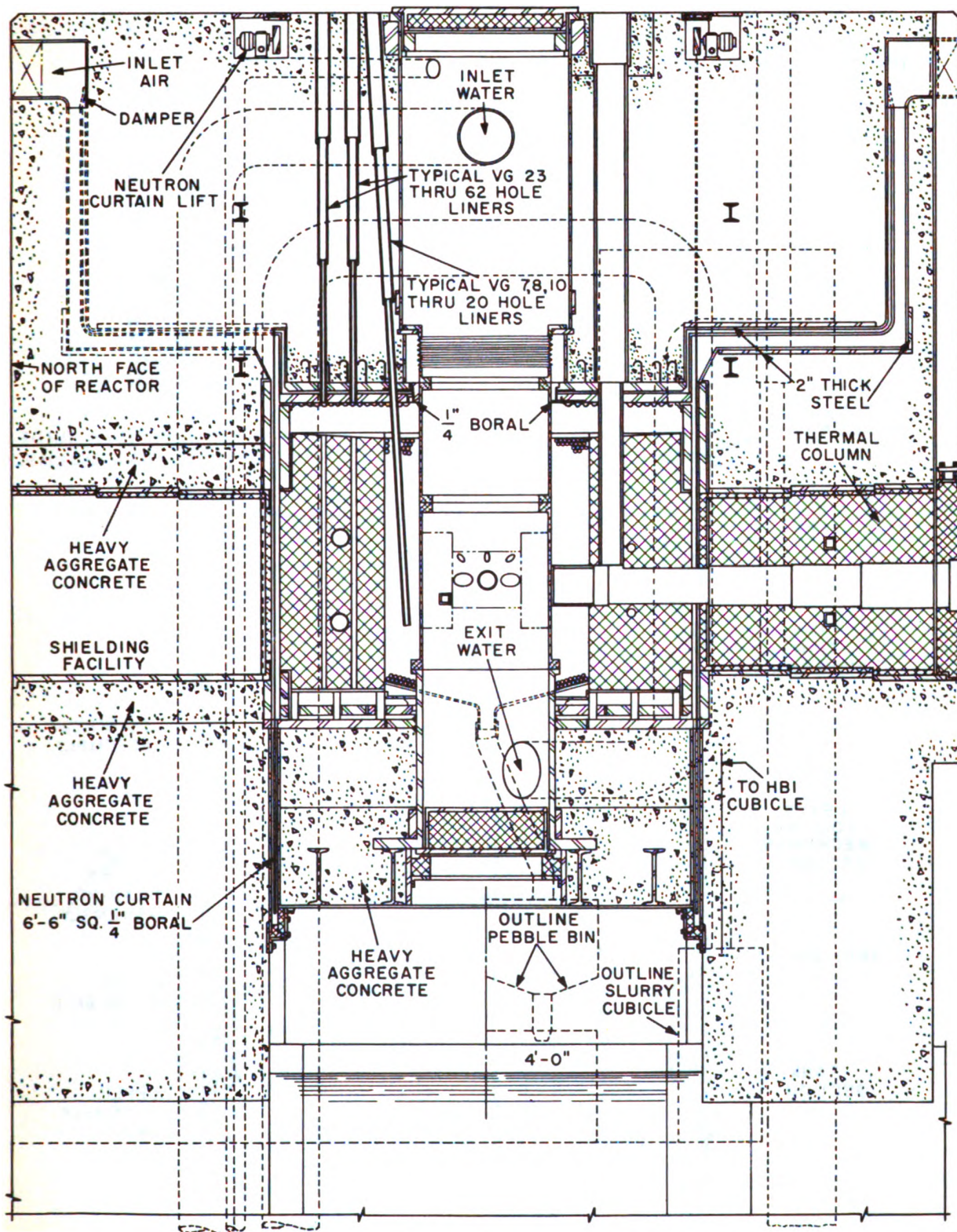


Fig. 3-79 Vertical section through north-south center line of reactor structure.

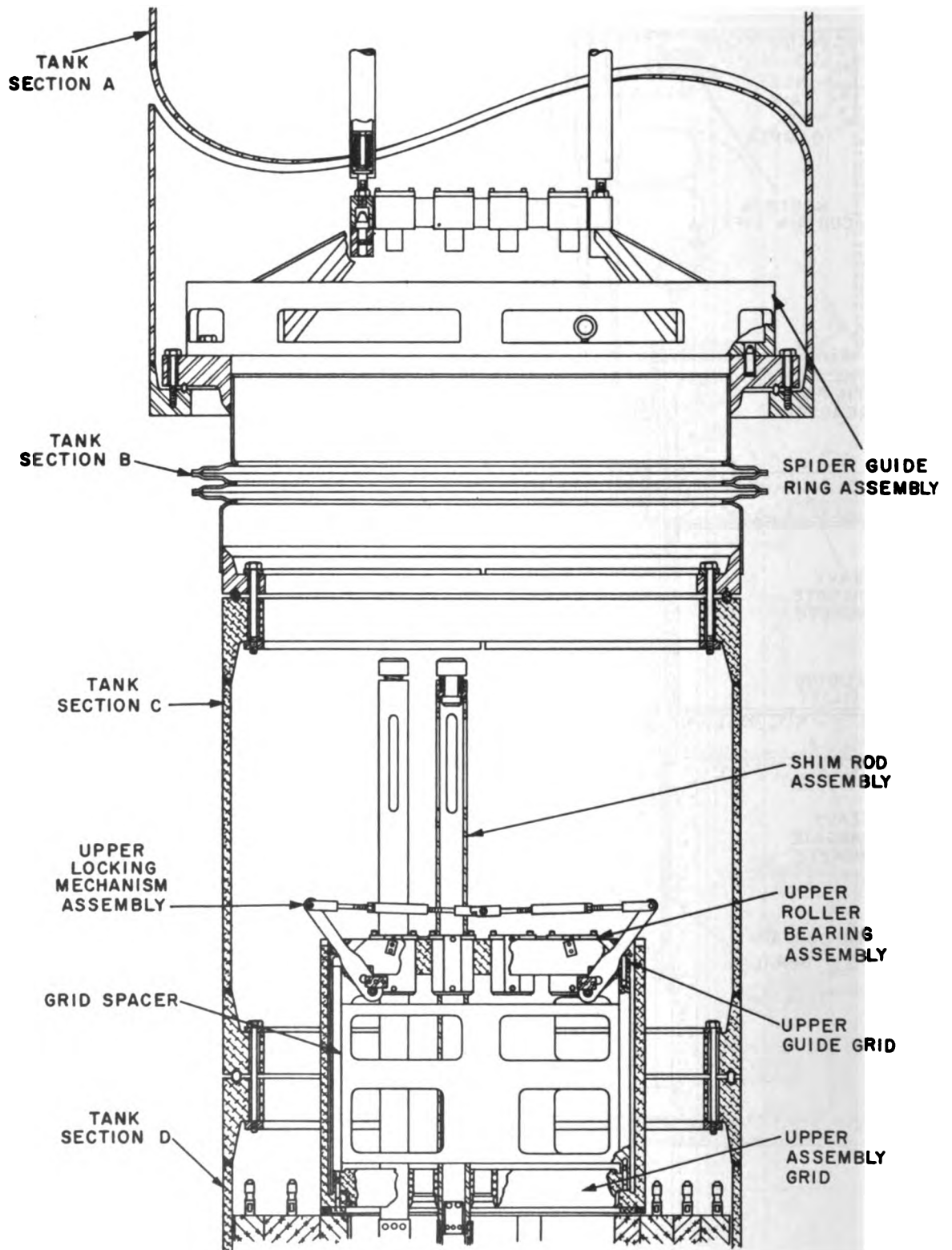


Fig. 3-80 Upper half of reactor assembly.

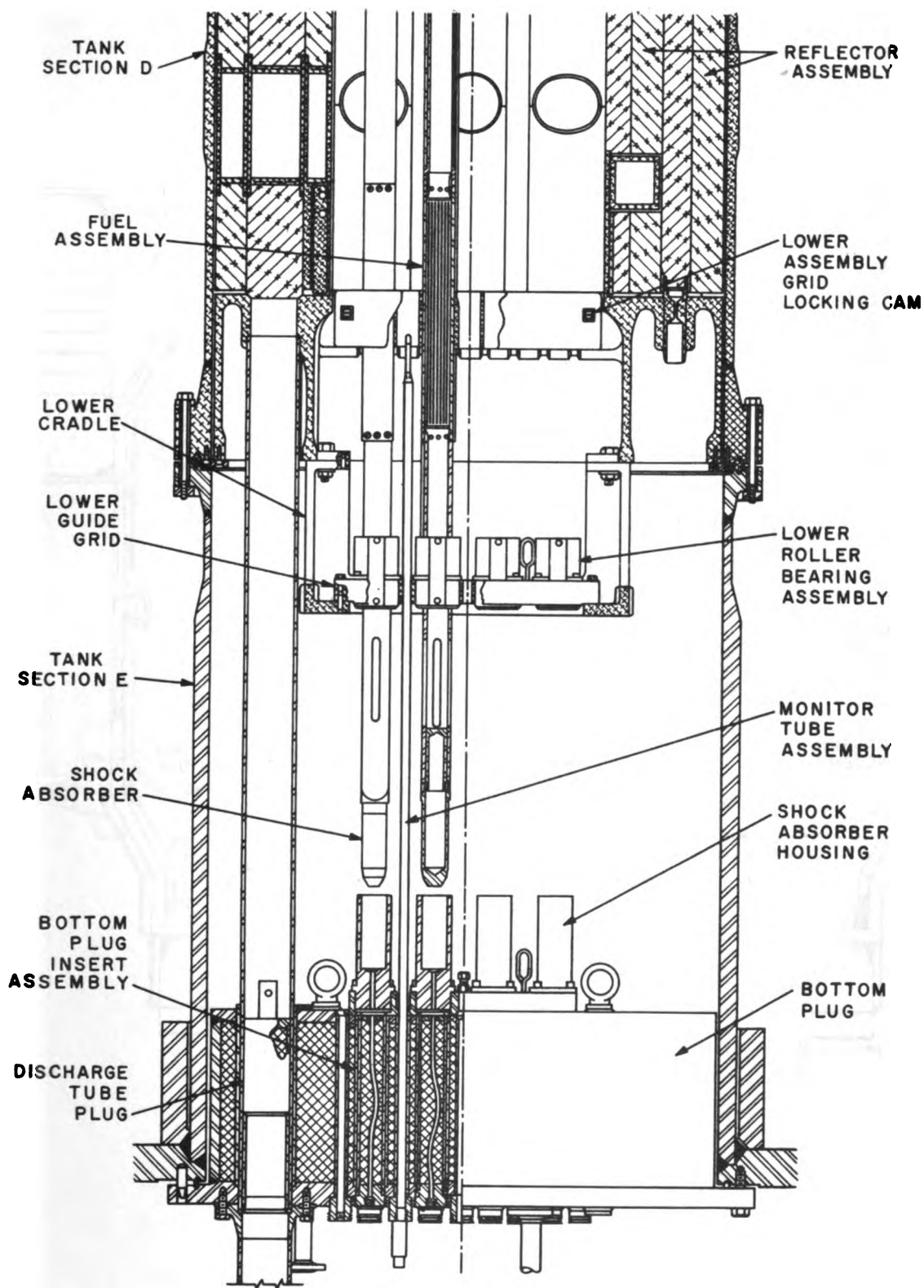


Fig. 3-81 Lower half of reactor assembly.

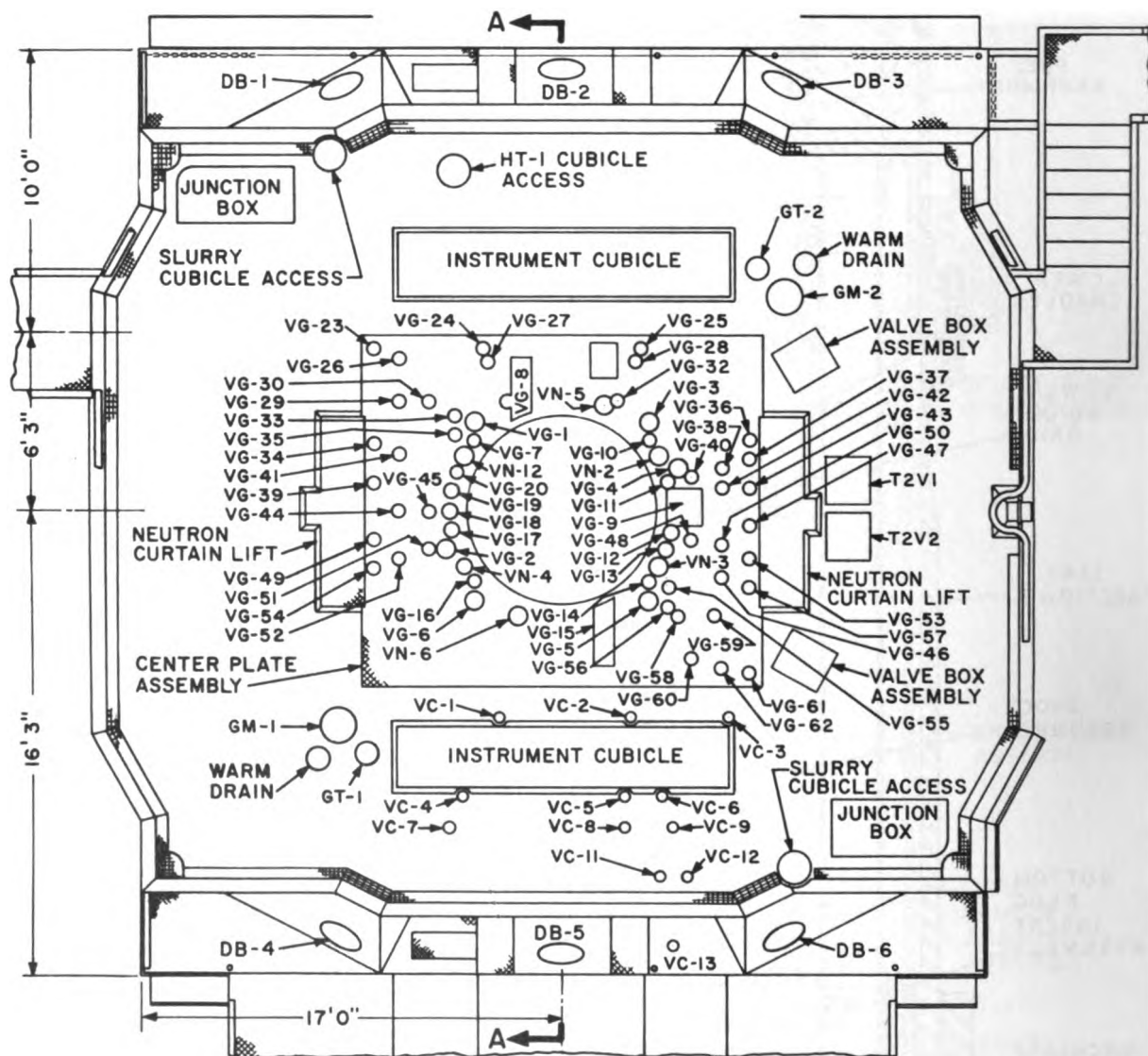


Fig. 3-82 Plan view of top of reactor.

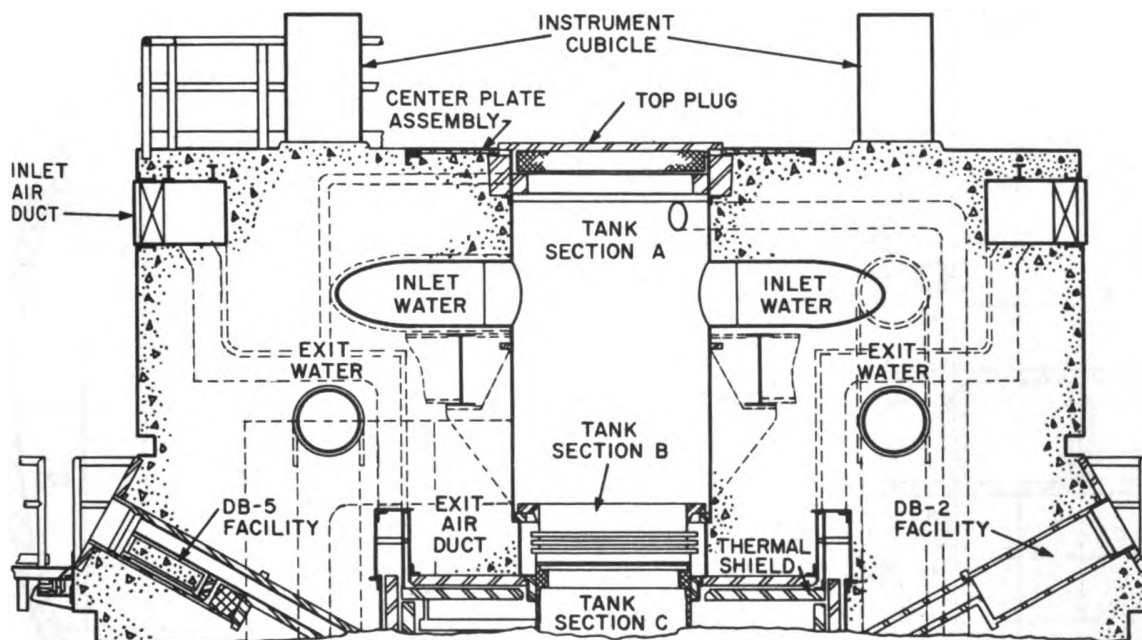


Fig. 3-83 Section A-A of Fig. 3-82.

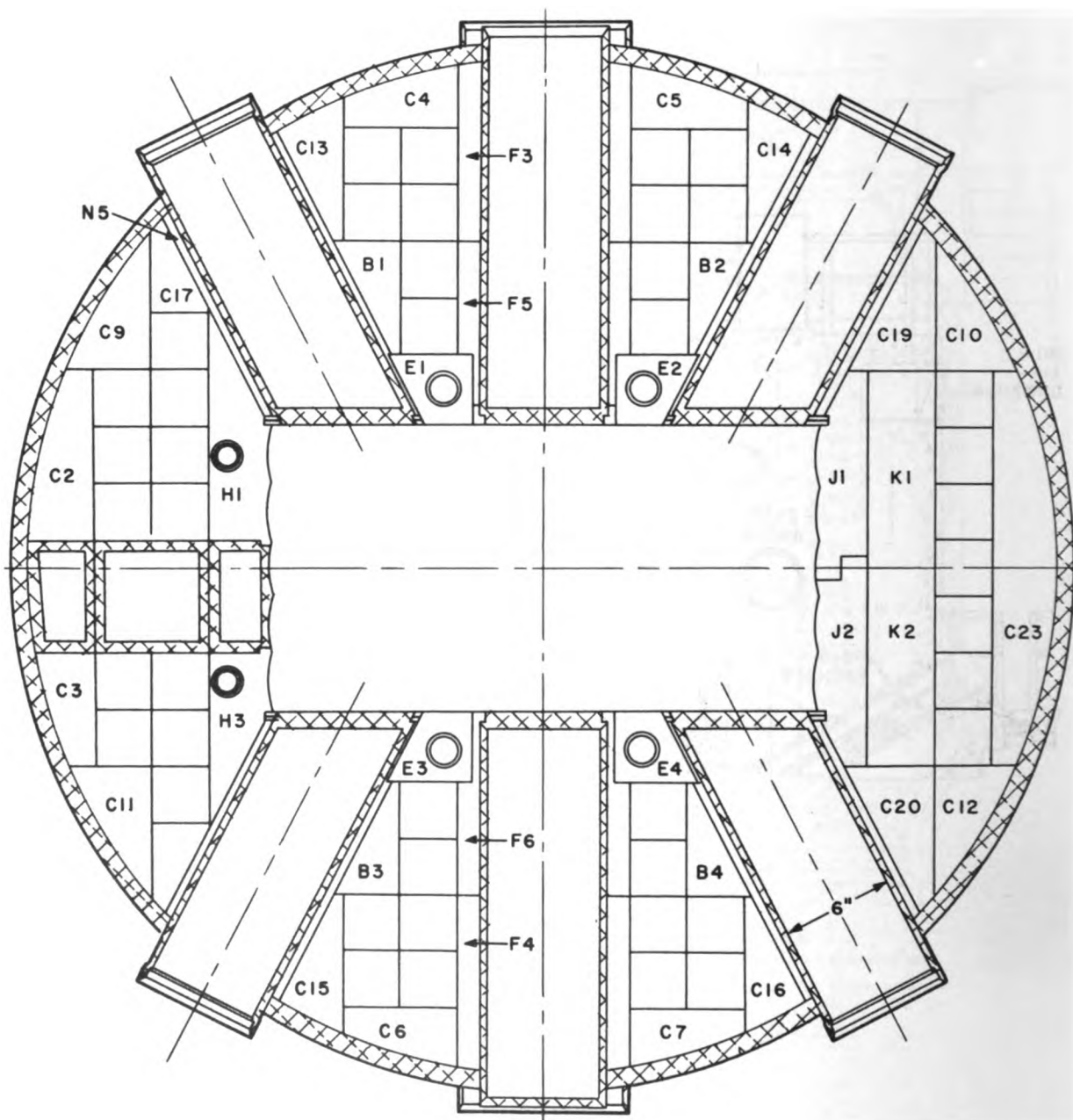


Fig. 3-84 Horizontal section through center line of reactor-tank reflector.

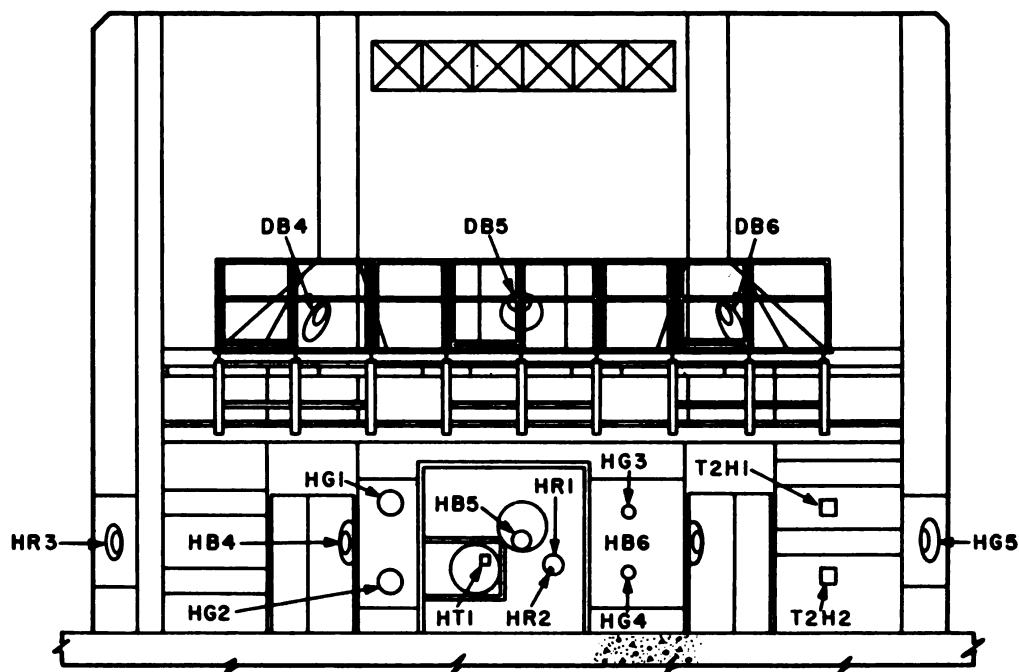


Fig. 3-85 West elevation of reactor.

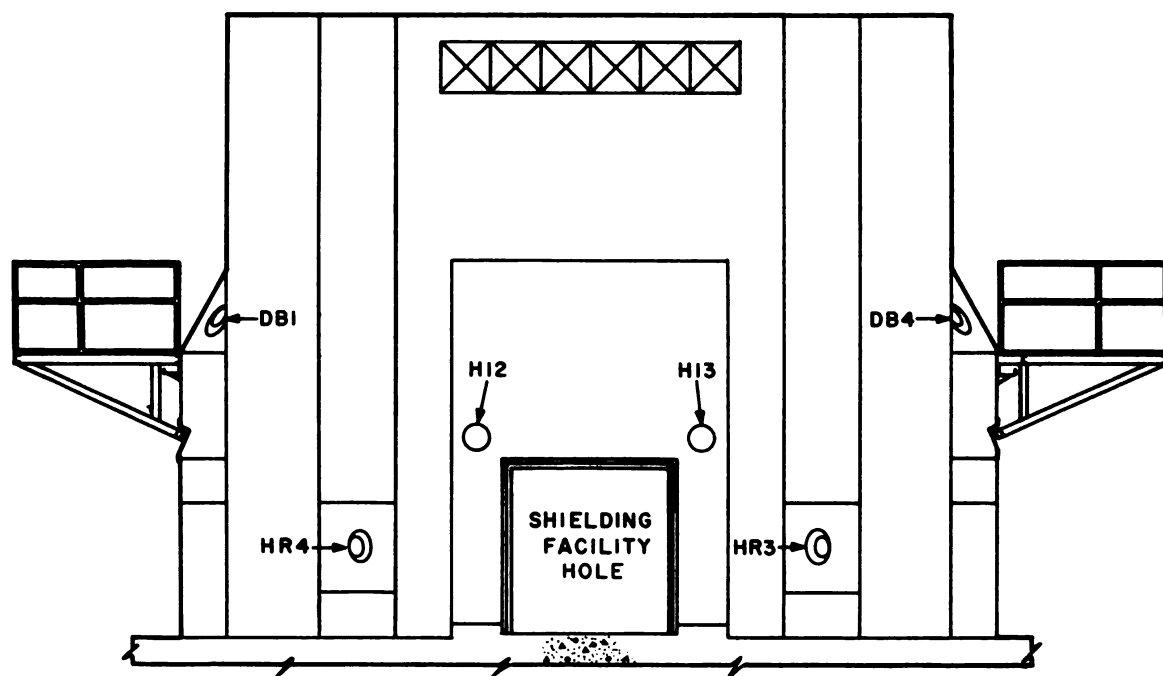


Fig. 3-86 North elevation of reactor.

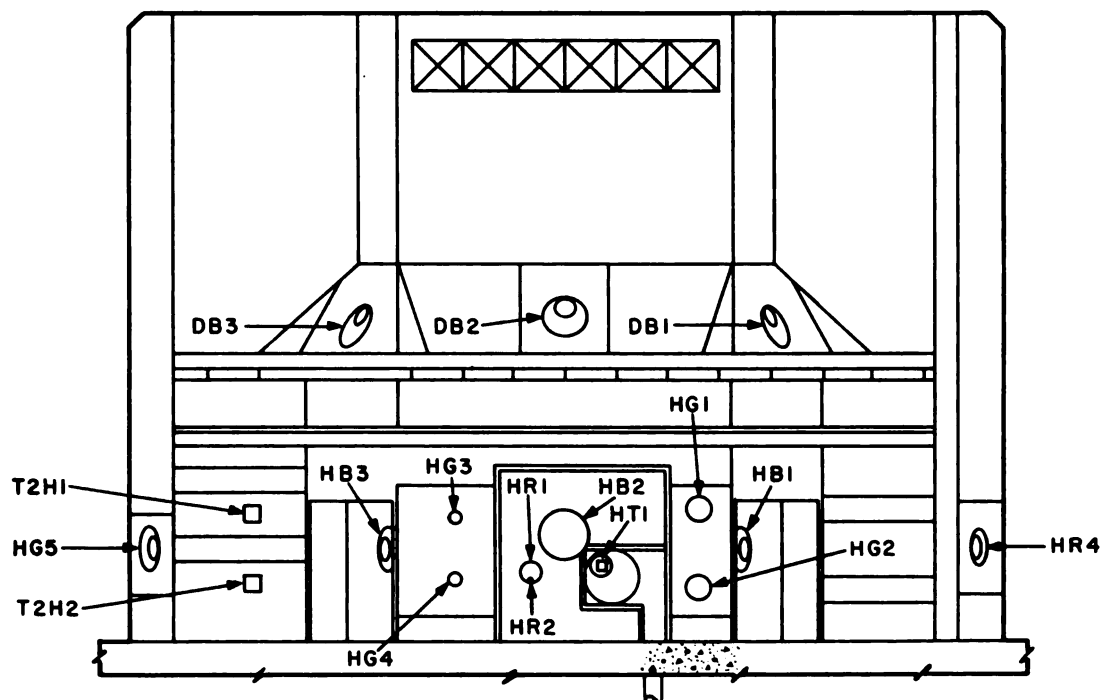


Fig. 3-87 East elevation of reactor.

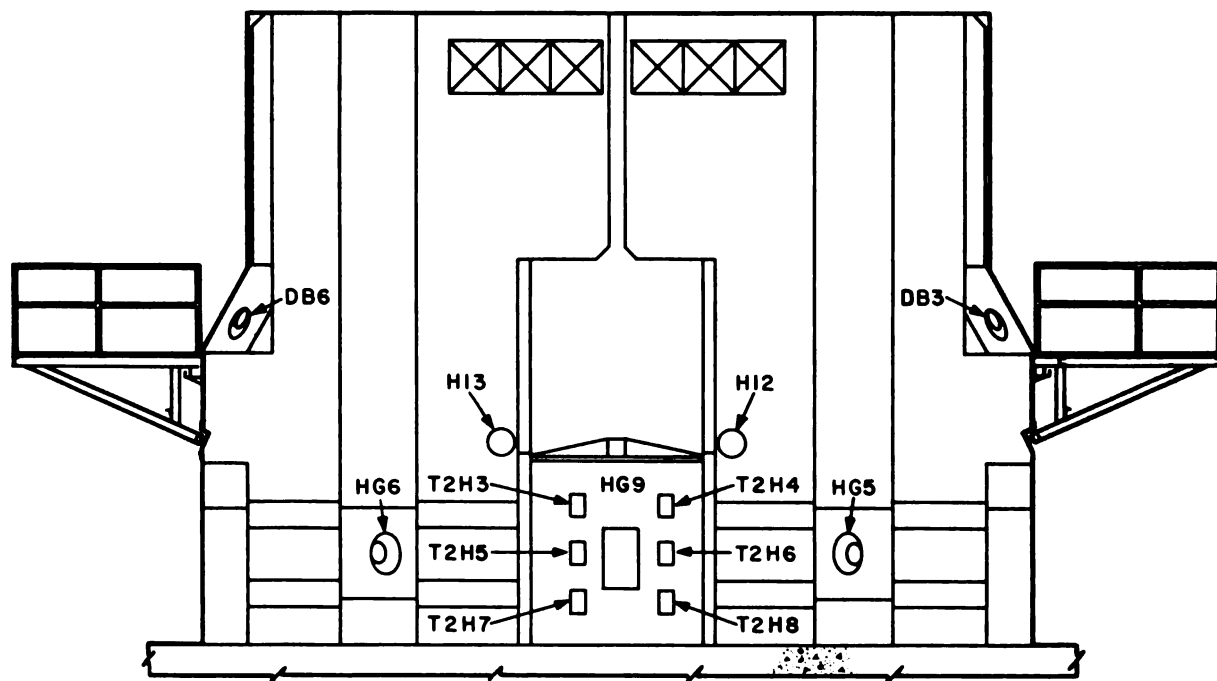


Fig. 3-88 South elevation of reactor.

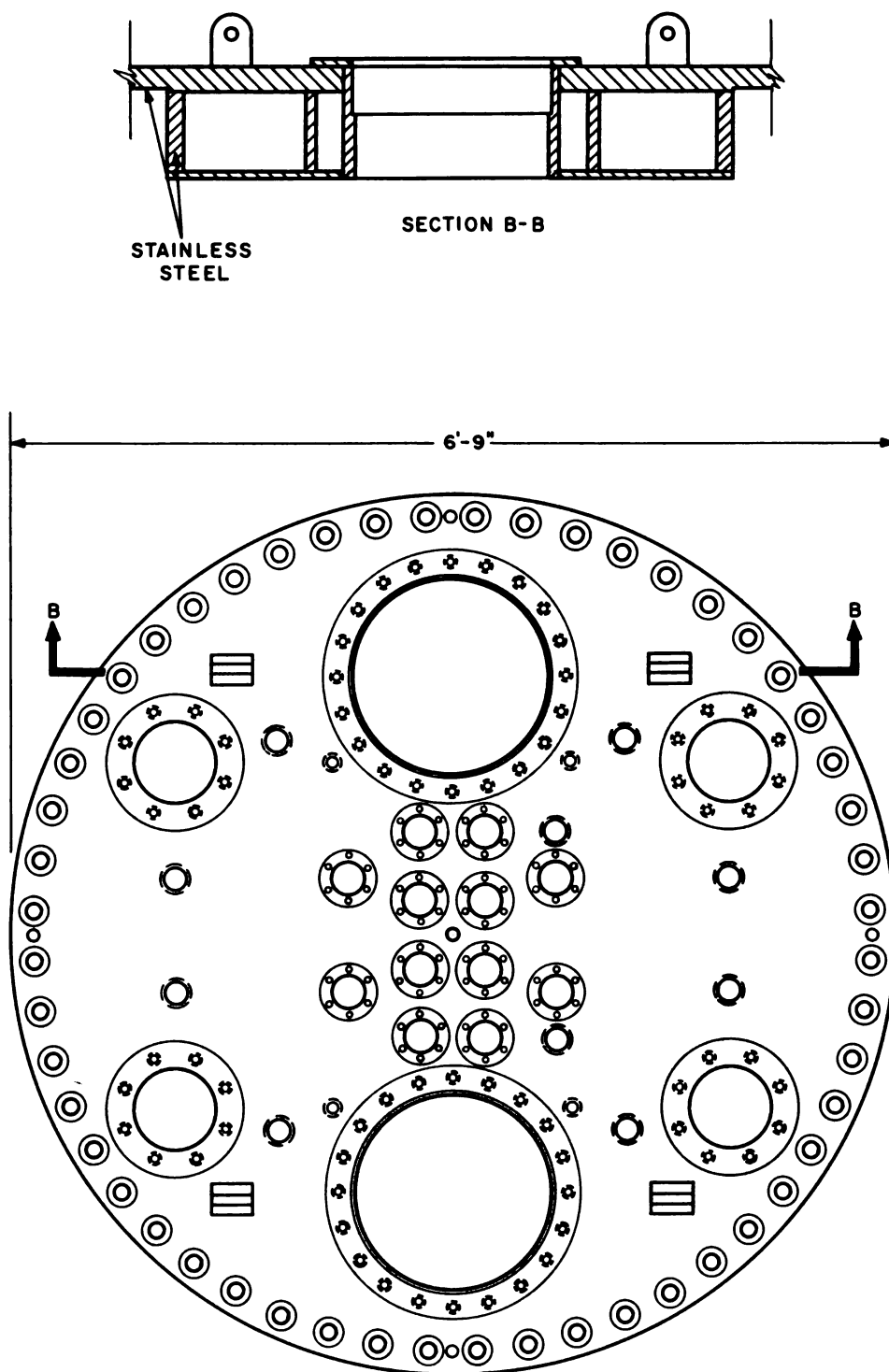


Fig. 3-89 Reactor-tank top plug.
299

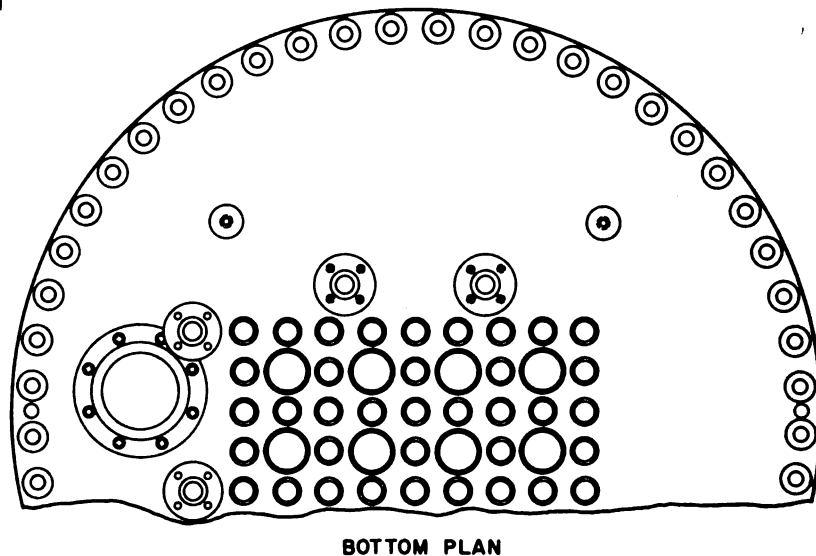
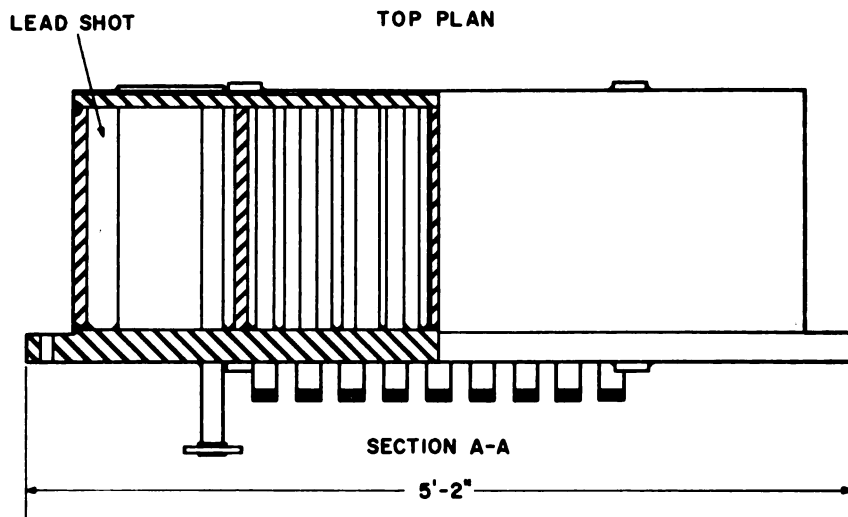
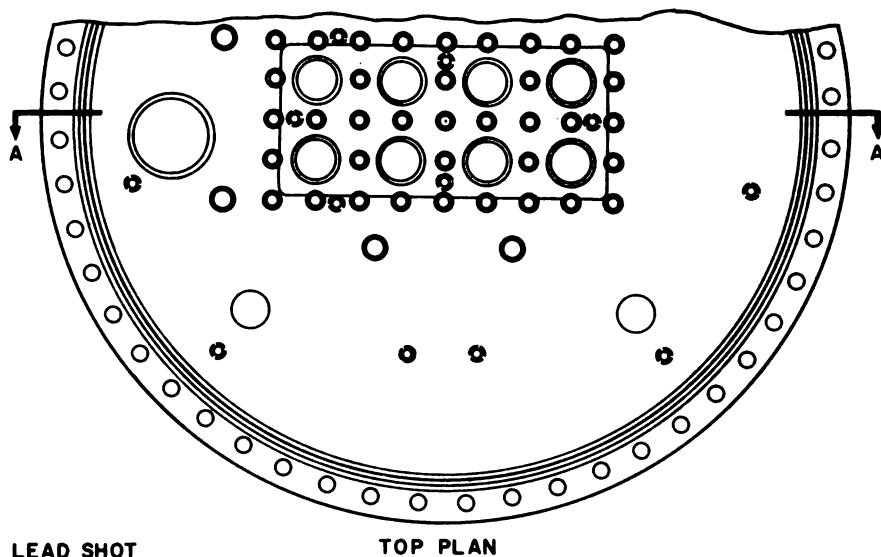


Fig. 3-90 Reactor-tank bottom plug.

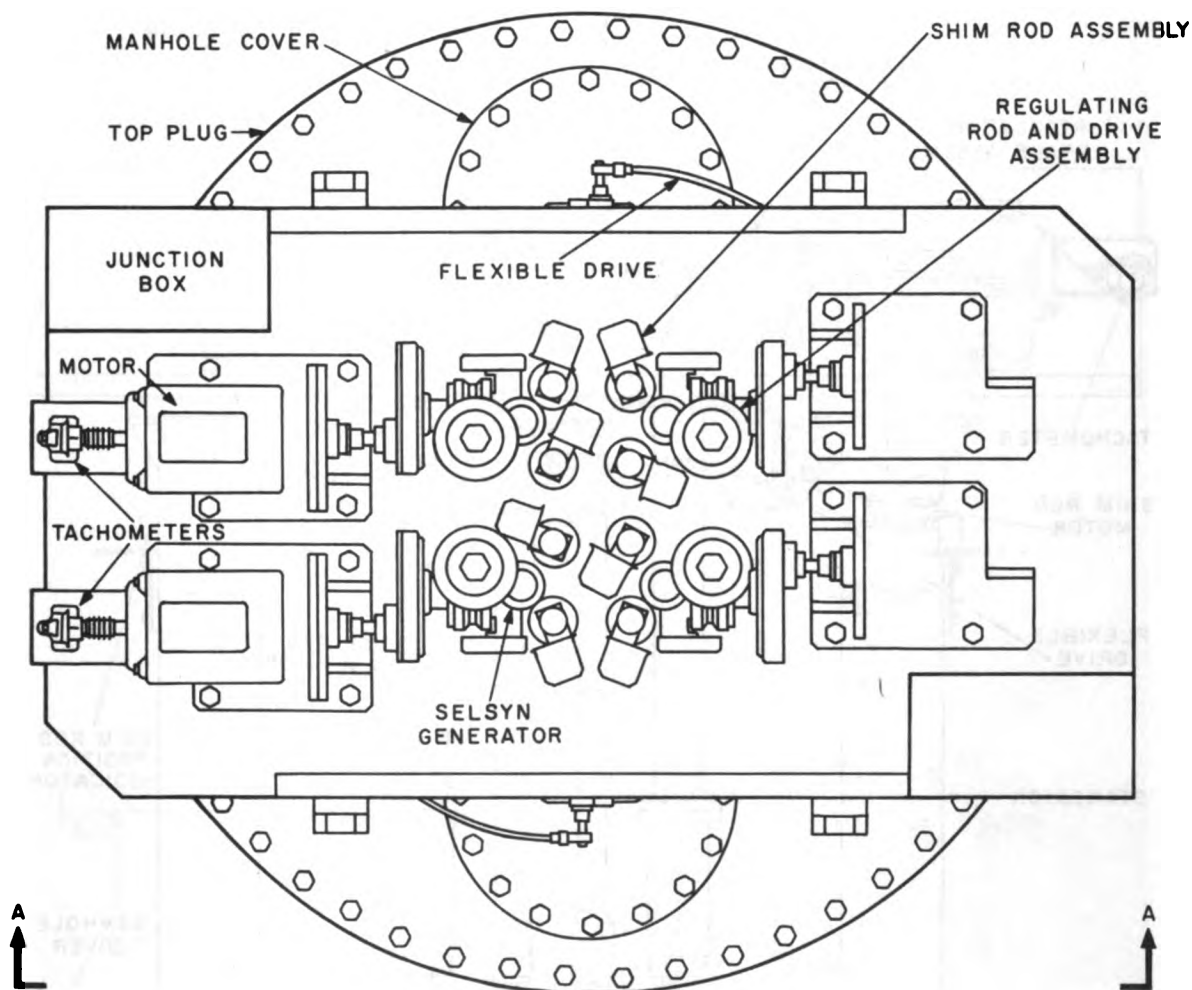


Fig. 3-91 Controls assembly, view I.

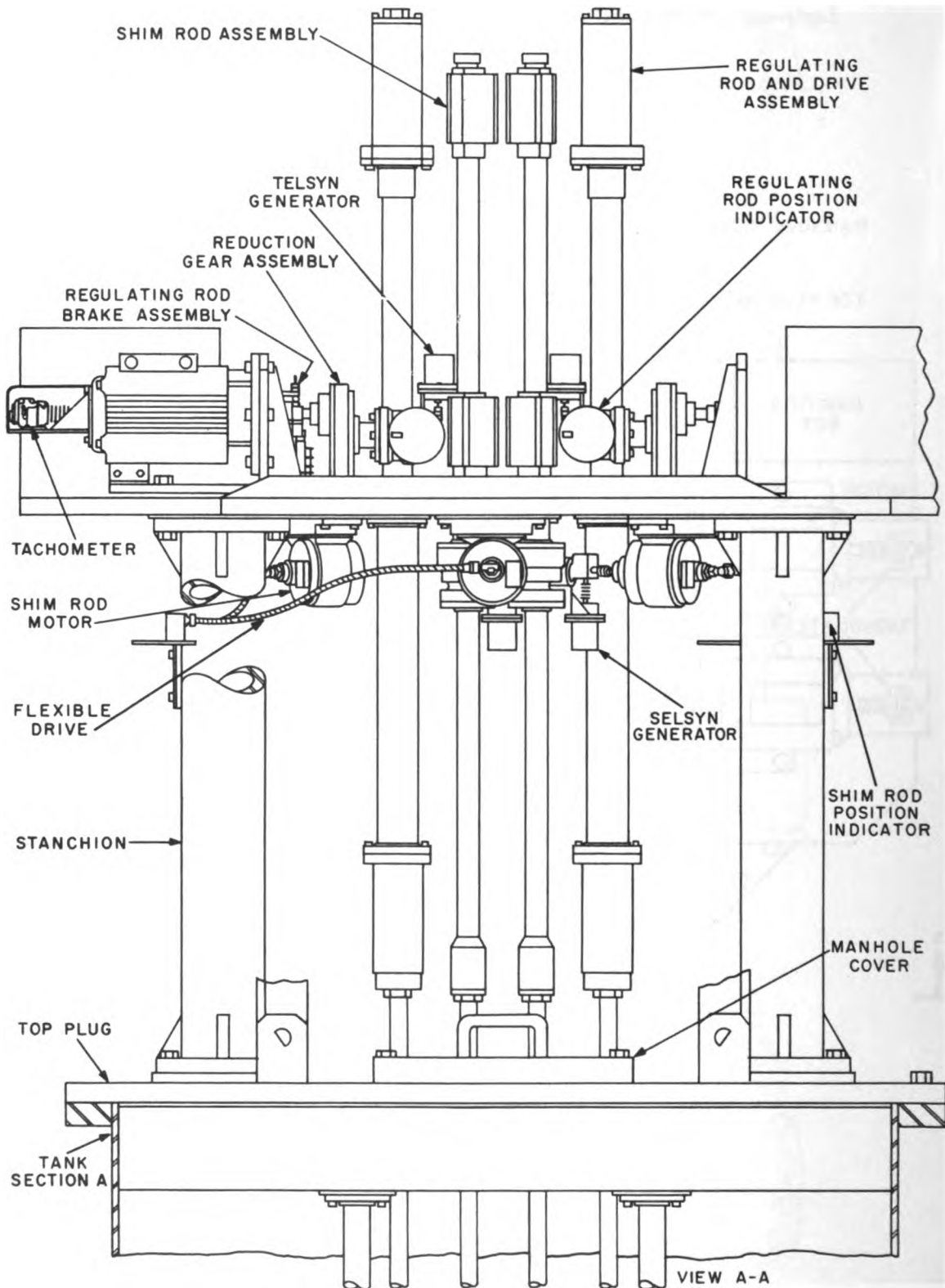


Fig. 3-92 Controls assembly, view II.
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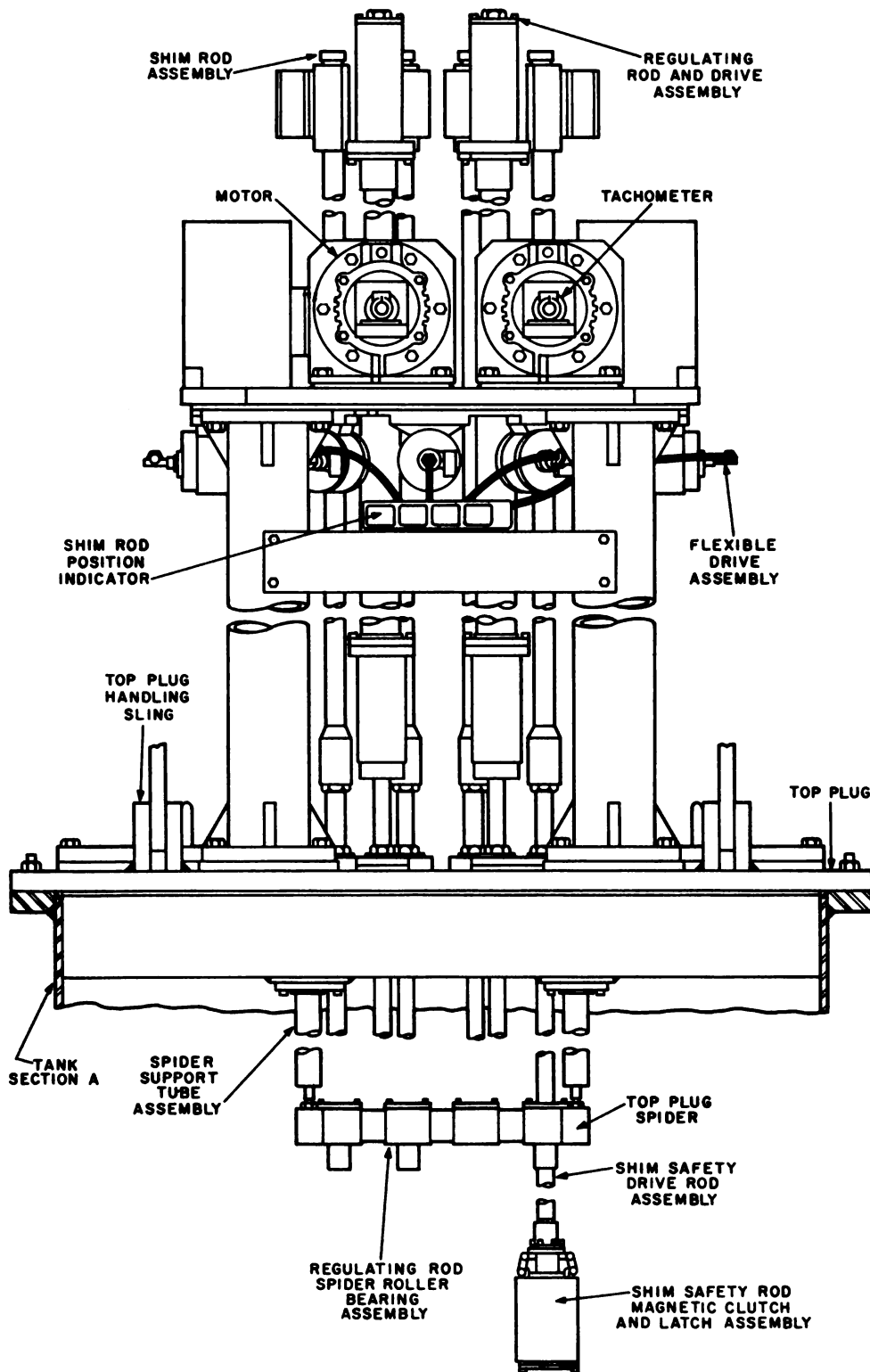


Fig. 3-93 Controls assembly, view III.

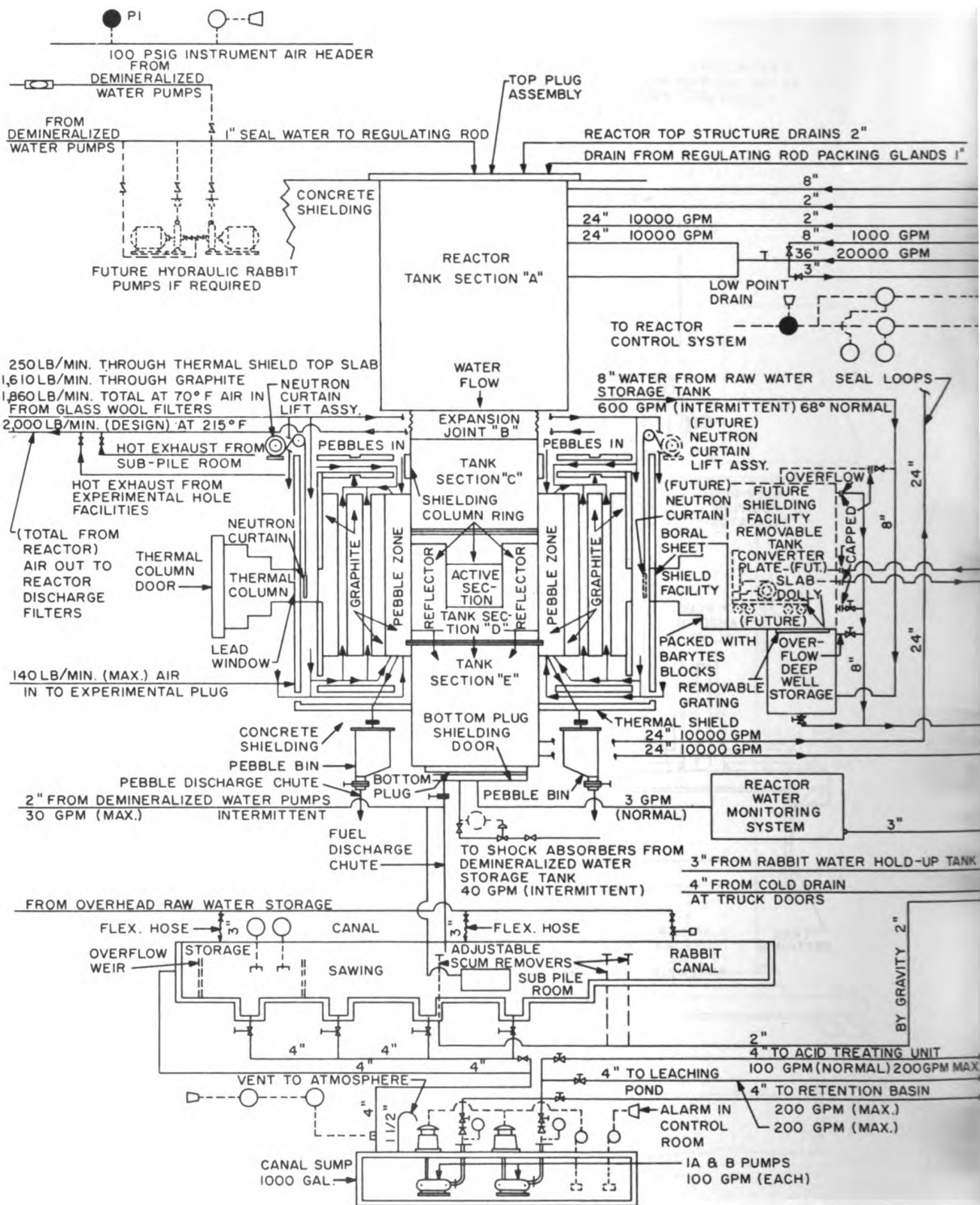


Fig. 3-94 Reactor-process system

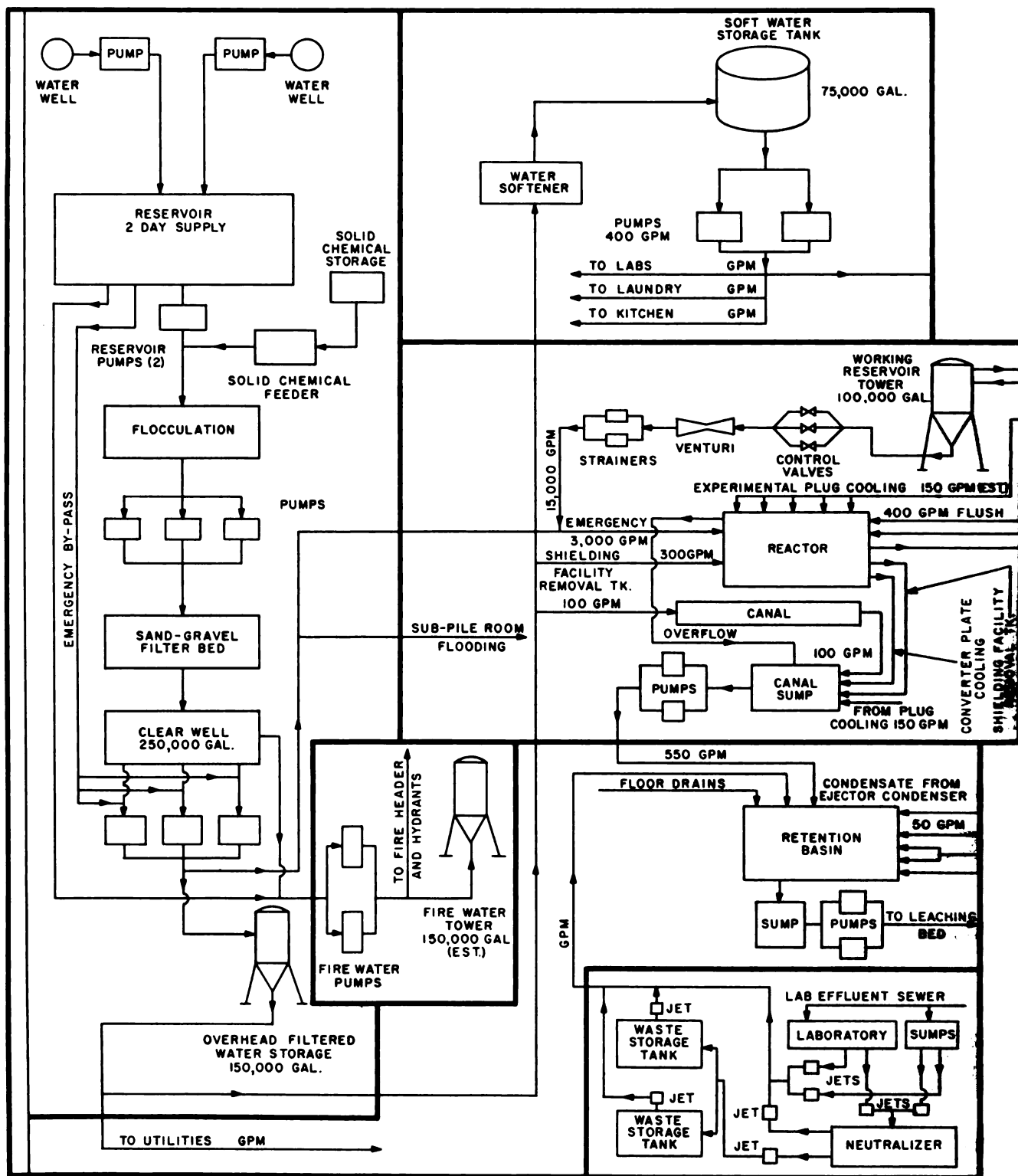
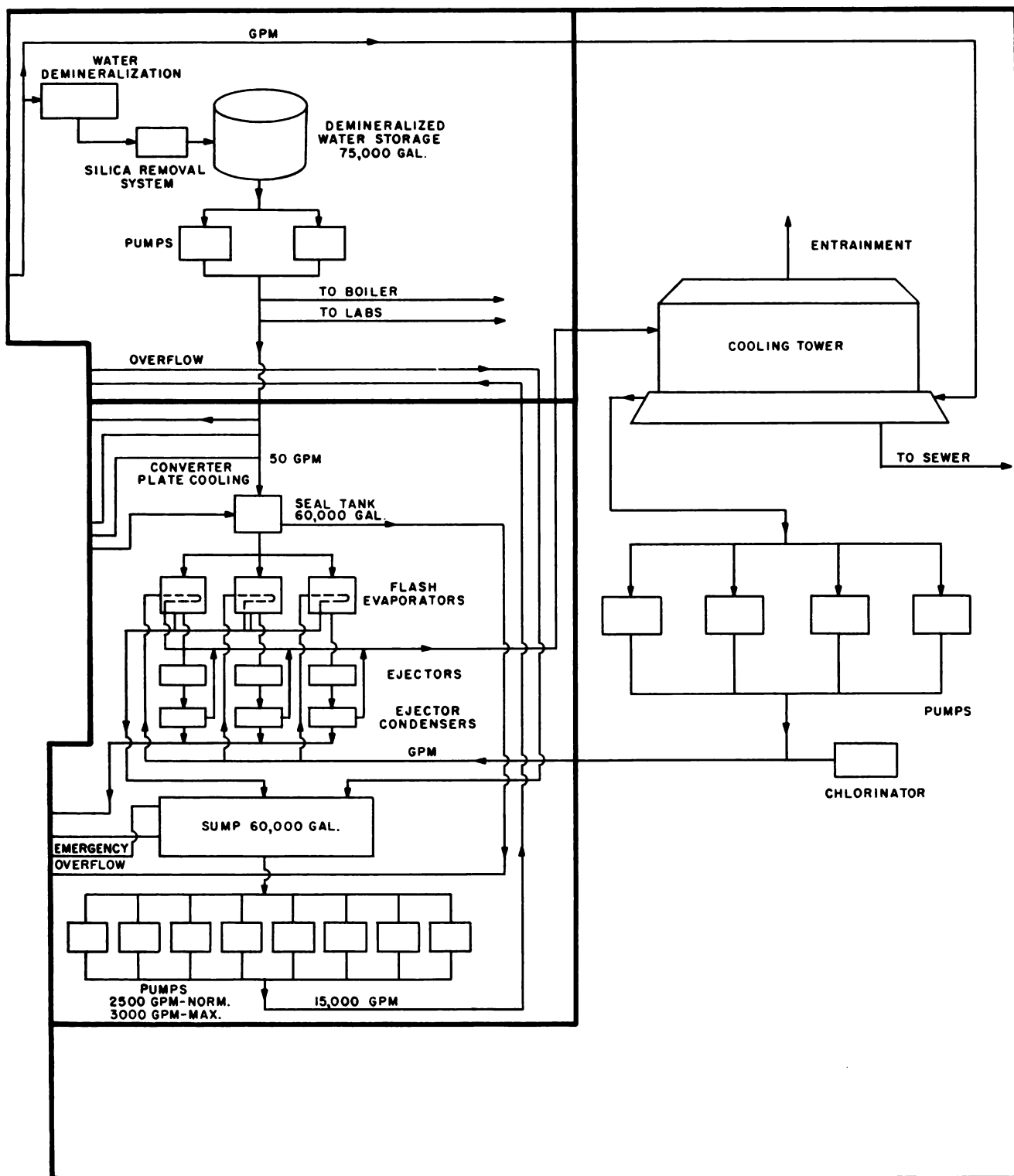


Fig. 3-95 Water-system



flow diagram.

CHAPTER 4

Light-water and Oil-moderated Reactor

Heterogeneous—Enriched Fuel

**COMPILED BY KNOLLS ATOMIC POWER LABORATORY
GENERAL ELECTRIC COMPANY**

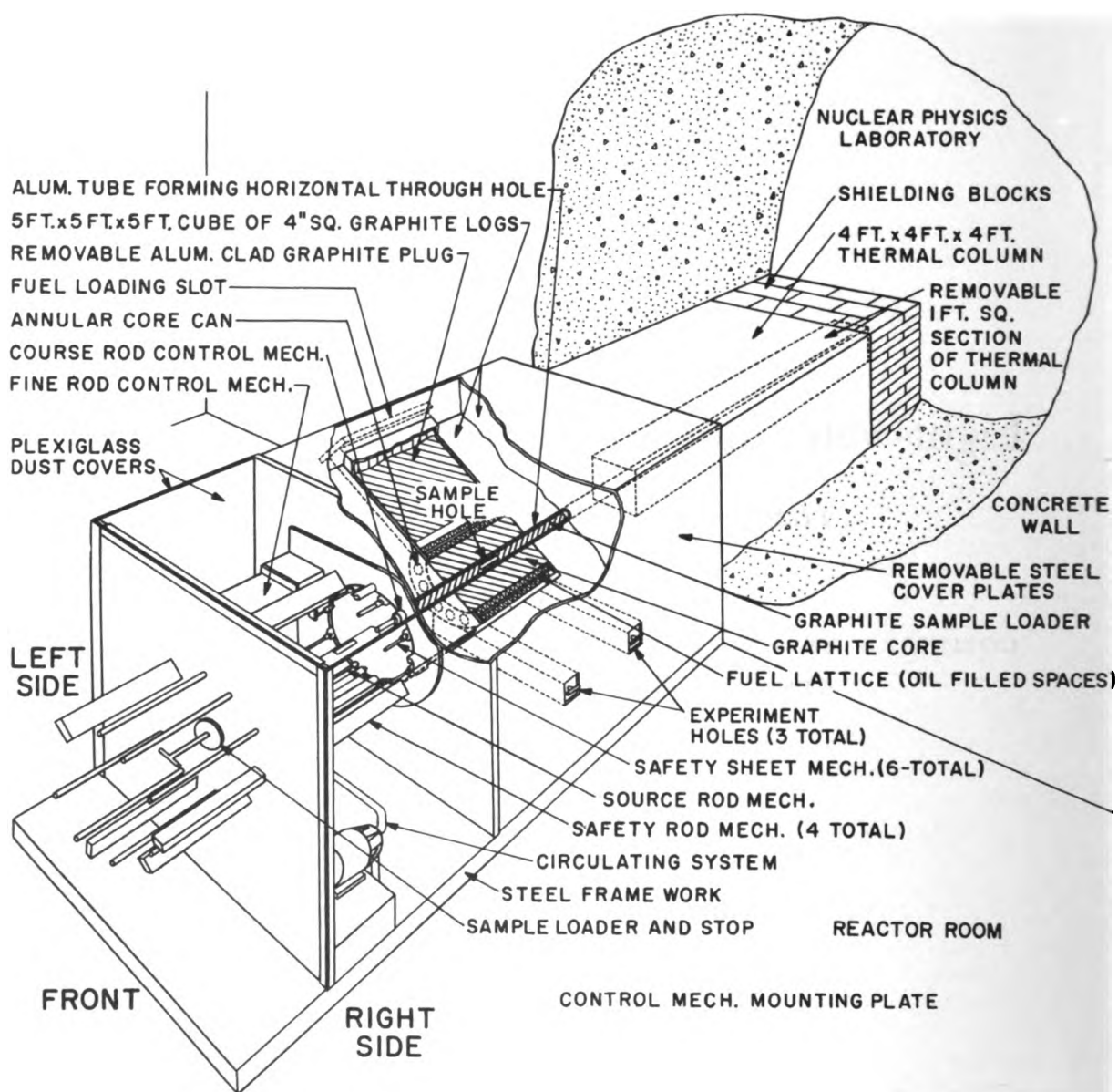


Fig. 4-1 Low-power nuclear test reactor, isometric.

CHAPTER 4

Light-water and Oil-moderated Reactor

Heterogeneous—Enriched Fuel

The nuclear test reactor (NTR) is the result of the logical evolution of a series of reactors built by a team of scientists at the Knolls Atomic Power Laboratory. The earlier reactors were known as thermal test reactors (TTR); three models have been built and operated successfully. The NTR and its predecessors were designed and constructed for experimentation in nuclear physics. Their capabilities were aimed in two directions: to provide a source of neutrons and to investigate the detection properties of a reactor.

The proficiency of nuclear reactors as neutron sources is a well recognized fact; however, their uses for detection of poisons or sources warrant some discussion. The control system of the NTR is extremely sensitive to changes in reactivity and is designed so that a minute change in reactivity can be measured. When a fissionable or poison sample is introduced into the reactor core the control rods must be moved to compensate for the positive or negative reactivity change. Therefore, the information of the efficiency of a sample as a source or sink of neutrons is presented in the form of changes in control-rod position. The position of the NTR control rod can be measured with 1 per cent accuracy, comparable to a change in reactivity of 5×10^{-5} .

The NTR is an enriched-uranium light-water and oil-moderated graphite-reflected water-cooled reactor. It is capable of operation at 30 kw. When operated at these higher powers, it provides a source of neutrons for its beam ports and ther-

mal column. At the center of the core, for the 30-kw unit, the thermal flux is 9×10^{11} neutrons/cm²/sec and fast flux is 2.25×10^{11} neutrons/cm²/sec. At low powers it becomes an analytical tool for studies using danger-coefficient methods or reactor-oscillator experiments. The proved high sensitivity and inherent safety of the earlier TTR models led to the redesign of the reactor in its present form for use as a versatile industrial- and nuclear-research tool.

Some of the possible uses of the NTR are listed below:

1. Nuclear-fuel quality control.
2. A standard neutron source.
3. Exponential and reactor mock-up experiments.
4. Danger-coefficient measurements.
5. Cross-section investigations.
6. Foil calibration.
7. Biological investigations.
8. Experimentation requiring collimated beams.
9. Food-preservation studies.
10. Heat-transfer studies.
11. Malignancy-treatment studies.
12. Heat-transfer studies.
13. Limited isotope production.
14. Personnel training.

The low-power installation of an NTR is shown in Fig. 4-2. An artist's concept of a high-power NTR installation is shown in Fig. 4-3. The total cost of a high-power NTR unit is about \$200,000, exclusive of shielding and building.

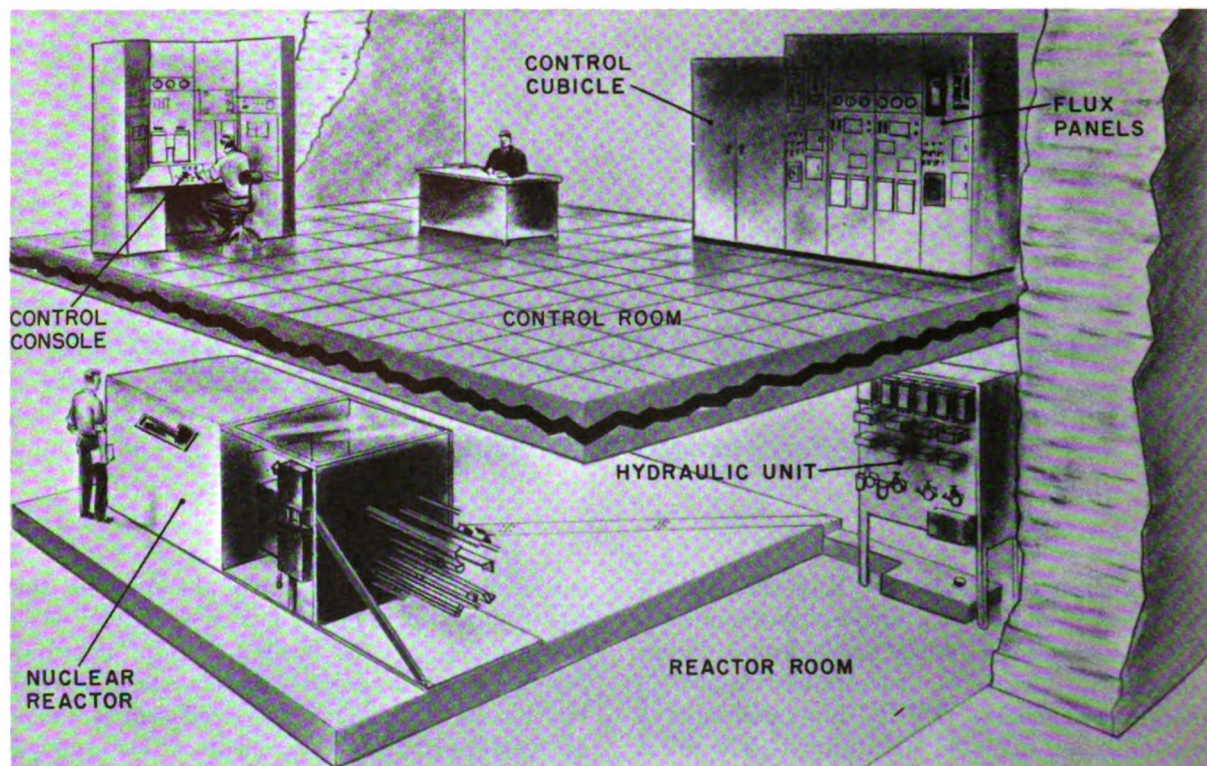


Fig. 4-2 Typical low-power installation.

GENERAL DESIGN FEATURES

The reactor consists of a core in the form of an annular cylinder centered in a 5-ft cube of graphite. The core itself is constructed of uranium-alloy disks on rods located in the annular aluminum container surrounding the central thermal column. The number of disks may be varied from the number required for activity as long as the excess is adequately compensated by control poison. The core cylinder is constructed of aluminum and is 19 in. long and 18 in. in diameter. At low power levels when no coolant flow is required, the fuel disks are immersed in paraffin-base oil or in light water. At power levels requiring coolant flow, water can be used.

With forced-water cooling, the NTR can be operated at power levels up to 30 kw. At this power level a laminar flow of about 25 gpm is sufficient to hold the temperature rise across the core to about 15°F. At the present time it is not known if turbulent flow will cause undesirable vibrations of the fuel disks. When this question

has been resolved, higher power levels may be possible. At power levels up to 30 kw, however, the reactor will provide a sufficient supply of neutrons to support many types of experiments using the thermal column, beam ports, and exposure holes.

The experimental facilities of this reactor are extremely versatile and several different facility arrangements can be made about the basic reactor. The thermal column is an extension of the graphite reflector and may be located on one of the three open sides of the reactor or may be extended upward, making it compatible with reactor mock-up studies using gravity-activated control rods. Another important experimental feature is the internal thermal column. This is a 12-in.-diameter internal graphite column with a 4-in.-diameter removable cylinder which can be loaded with small samples. Several beam ports can be located radially about the core, extending outward through the biological shield and terminating with automatic beam-port shutters.

Low-power operation of the reactor, which does

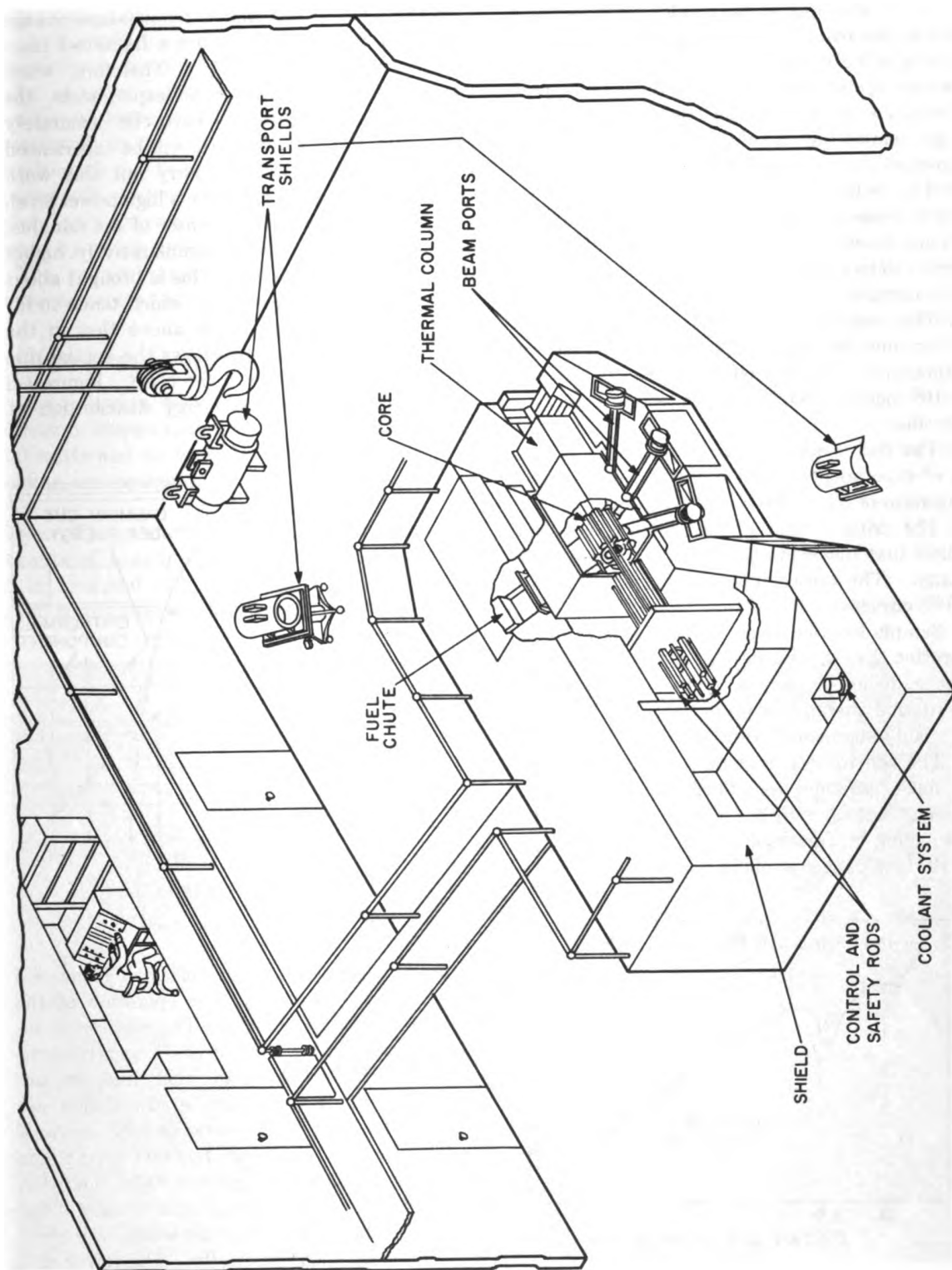


Fig. 4-3 Typical high-power installation.

not require shielding of the reactor itself, permits access to the reactor during shutdown and allows handling of fuel elements after a short decay time. However, operation from 100 watts to 30 kw precludes any such freedoms and necessitates shielding and remote fuel handling. At the 30-kw level approximately 5 ft of concrete shielding is required to reduce the gamma, thermal, and fast-neutron fluxes to a value that permits safe access to points outside the shield. For application as a sensitive detector, the reactor meets the following design criteria:

1. The neutron flux per watt at the sample position must be large. This is accomplished by an annular core that produces a thermal flux of 3×10^7 neutrons/cm²/sec per watt along its center line.

2. The flux spectrum should allow interpretation of experiments. The NTR provides a cadmium ratio of 5.9 at the sample position.

3. The critical mass of the reactor must be small so that the fractional reactivity change will be large. The critical mass of the NTR is 2.5 kg U²³⁵ enriched.

4. Sample location must be such that neutron absorption by the sample is very effective in either reducing or increasing reactor reactivity. The internal thermal column of the NTR has an appreciably augmented neutron flux.

5. The control system must be very sensitive and must provide reproducibility. This requirement is met with a control system capable of detecting a fractional reactivity change of 5×10^{-7} by danger-coefficient methods.

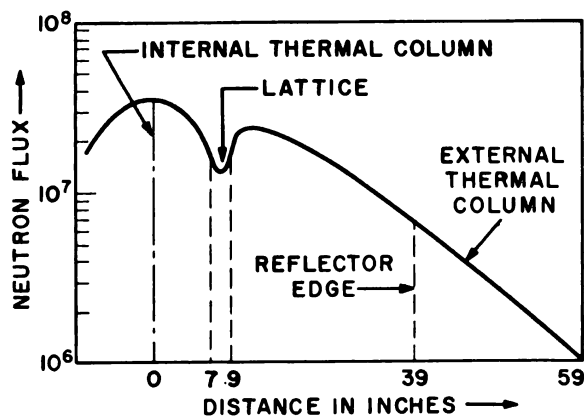


Fig. 4-4 Flux plot.

It should be noted that a temperature change of $5 \times 10^{-3}^{\circ}\text{C}$ will also induce a fractional reactivity change of 5×10^{-7} . Therefore, when carrying out precise detection experiments, the reactor-room temperature must be accurately regulated. Similar difficulties will be experienced if an attempt is made to carry out this work immediately after operating at a high power level.

The thermal flux at the center of the annulus, discussed in (1) above, is comparatively higher than most other reactors. This is brought about by the unique core geometry, which tends to increase the flux in the center above that in the active lattice. Figure 4-4 shows the spatial distribution of thermal flux in the NTR. Figure 4-5 is a plot of the neutron-energy distribution at the sample position.

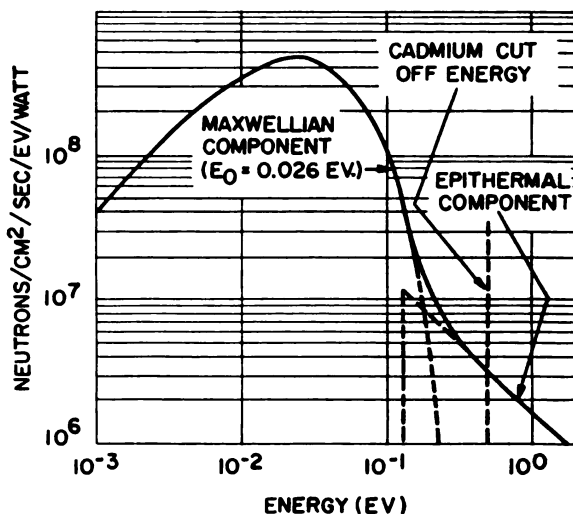


Fig. 4-5 NTR neutron spectrum.

The probability that a nuclear incident will occur as a result of improper operation of the NTR is extremely remote. The reactor is inherently self-limiting because of its negative temperature coefficient when the fuel disks are immersed in oil. Incorporated in the design are several "fail-safe" automatic safety devices. Their action is governed by monitoring the neutron-flux level, the gamma level, and the power period. When power fails or any other potentially dangerous situation exists, the reactor is shut down automatically. The safety and control functions of this design are completely

separated. The control functions are carried out by two coarse rods and one fine rod. Safety functions are carried out by four spring-actuated rods. Another safety feature is the inclusion of shutdown sheets which provide sufficient negative reactivity to preclude any possible danger of criticality due to an error in fuel loading. The coarse rods are pneumatically operated and the safety rods are removed pneumatically and inserted by spring force.

The reactor instrumentation is of standard research-reactor type.

DESCRIPTION OF THE REACTOR

The following description applies to this type of reactor except that, for operation greater than 100 watts and up to 30 kw, shielding and forced cooling are required.

Core. The reactor core, as shown in Fig. 4-6, consists of a cylindrical annular aluminum can 19 in. long and 18 in. in diameter with an inside annulus diameter of 12 in. Attached to the side

of the can and inclined at an angle of about 40° is a 3- by 19-in. rectangular aluminum loading chute used as a guide for loading the reactor. The entire unit is seam-welded at the edges with a joint that can be ground off and rewelded easily when the internal core condition must be evaluated. The sides of the core consist of $\frac{1}{2}$ -in.-thick aluminum plates machined to the proper dimensions and configuration. Machined on the inside surfaces of these plates is a pair of circular raceways which terminate at the loading chute. Milled slots down the sides of the loading chute meet these circular raceways. The raceways serve to guide the loaded reel assembly.

Welded to the outer periphery of the core are eight $\frac{1}{2}$ -in. aluminum tubes which project out from the reactor can to the control face of the reactor. These tubes are used to guide the control rods, the safety rods, and the source rod.

The outer periphery of the core assembly is covered by $\frac{1}{16}$ -in. aluminum sheet welded to the core plates at the specially designed edges. Before assembly in the graphite pack, the entire core is helium-leak checked and then pressure checked.

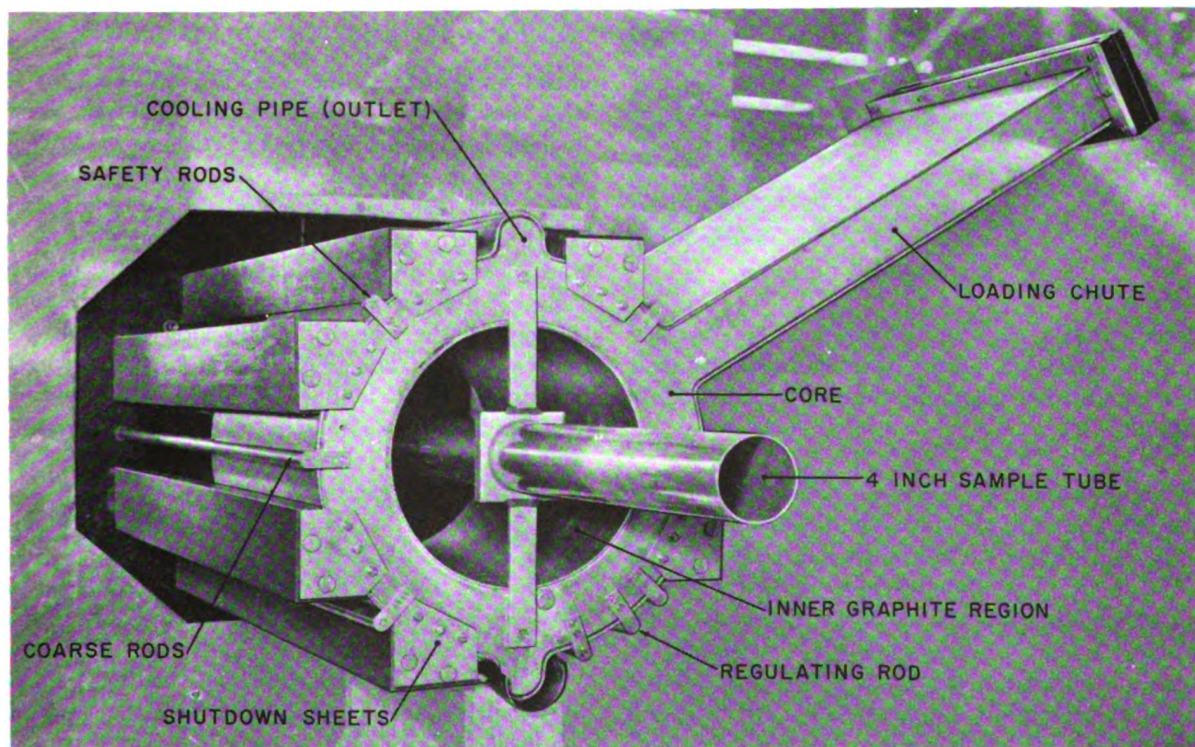


Fig. 4-6 Core assembly, rear view.

Incorporated into the basic design of the core is a pair of 2-in.-diameter inlet and outlet pipes used for forced cooling of the core (Fig. 4-7). The inlet pipe has a diverter so located as to provide equal flows of coolant to both sides of the core.

Two vent tubes are located on the core assembly for bleeding the system of any gases which may form during operation. The coolants are so

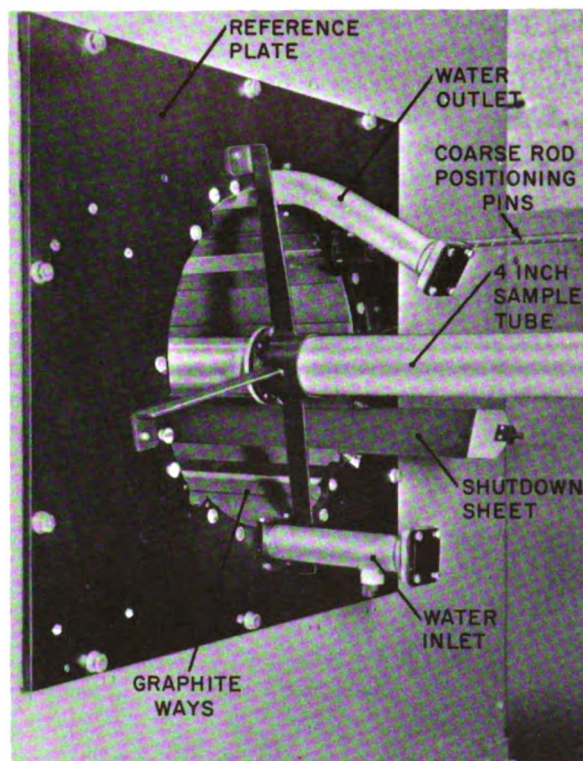


Fig. 4-7 Core assembly, front view.

chosen that minimum gas evolution is obtained during operation under neutron bombardment. The entire core assembly weighs approximately 200 lb.

Main Graphite Moderator. The reactor core assembly is located within the center of the 5-ft cube of boron-free high-purity graphite. The main moderator and reflector is composed of approximately 285 blocks of graphite logs 4 in. square and 48 in. long, stacked in a manner to obtain a minimum of voids. Since the friction between machined surfaces is slight, a small amount of binder is applied to the horizontal surfaces to prevent slipping during stacking.

The graphite blocks totally enclose the reactor core and serve to moderate and reflect the outgoing neutrons to reinforce the reaction within the reactor. The graphite reflector is covered on all sides except the front with a sheet of $\frac{1}{32}$ -in.-thick sheet cadmium, which is used to capture the thermalized neutrons. The cadmium lining is overlaid with a $\frac{1}{4}$ -in.-thick boiler plate. The boiler plate serves primarily to hold the pack together but also provides slight gamma shielding.

Experimentation holes are provided which run normal to the axis of the reactor loading tube. These holes extend from one side of the reactor to the center of the graphite pack. In order to maintain the neutron loss at a minimum and to conserve fuel, unused openings are completely plugged with graphite.

The surfaces of the graphite pack which come in contact with the core assembly are machined to fit snugly around the core. The guide channels for the shutdown sheets are also machined into the graphite so that the graphite acts as its own guide and lubricant.

The main graphite moderator weighs approximately 7 tons.

Fuel Assembly. Figure 4-8 illustrates a typical fuel assembly. A single fuel assembly fully loaded for critical conditions consists of thirty-five 2.75-in.-OD and 0.375-in.-ID by 0.06-in.-thick fuel disks assembled on an aluminum rod 18 in. long. The spacer material for separating the disks is aluminum pipe cut to the proper length. Sixteen of these fuel assemblies can completely fill the reel within the core to create the active lattice. The 16 fuel assemblies are at present designed to hold a total of approximately 3 kg of fuel. An alteration of the spacer dimension can be made to permit the use of a maximum of 60 disks per assembly. The total amount of fuel which can be loaded into the reactor under these altered conditions is 6 kg.

The fuel disks in the assembly are immersed in General Electric 10C oil or in light water, which fills the annular space of the core.

A manually manipulated fuel-loading tool is provided for loading and unloading the reactor. As shown in Fig. 4-8, it is rectangular in construction and is provided with controlled spring clips

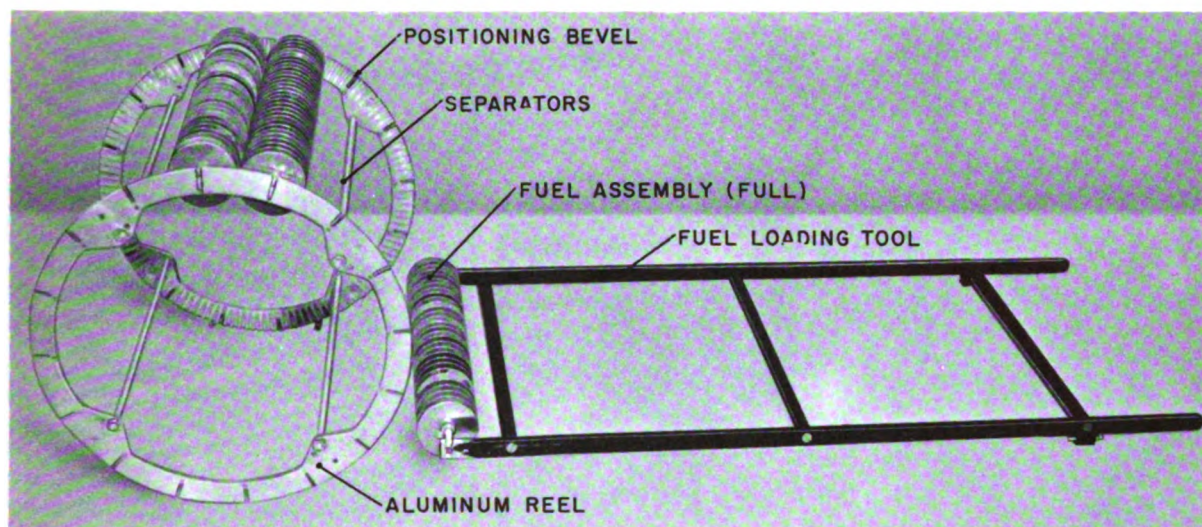


Fig. 4-8 Fuel-reel assembly.

at the loading side to allow the loading and unloading of moderately radioactive fuel.

Reel Assembly. Located within the welded reactor can is a reel assembly (shown in Fig. 4-8) constructed of a pair of ring gears tied together by a number of aluminum separators. This reel is externally positioned and controlled by a knob at the chute end which rotates the entire reel assembly through a pinion gear.

A fuel assembly ready for insertion into the reactor is first loaded into the chute slots and is then guided with a loading tool down these slots to the reel assembly. If the reel is not positioned properly, the loaded fuel assembly rests against the outer ring edges. By rotating the position knob, the reel can be turned until a pair of notches cut into the rings lines up with the milled chute slots. The fuel assembly then slides into the notches, automatically placing the fuel assembly into the circular raceway. Further rotation of the knob turns the reel with its seated fuel assembly.

An external dial indicator geared to the drive knob (shown in Fig. 4-9) continuously reveals to the operator the position of any fuel assembly within the reel. A written record is also kept of the number of disks per assembly and their relative positions in each assembly.

After the reactor is loaded with fuel, the loading chute is plugged with an aluminum-clad graphite block and the opening covered with a

gasketed aluminum plate to prevent oil or water leakage. The design of this cover plate provides for quick release of any pressure build-ups.

Central Thermal Column. An annular cylindrical form is constructed from graphite having an outside diameter of approximately 12 in. and an inside hole dimension of about 4 in. The graphite column is located in the central hole of the aluminum core. It functions to moderate the neutrons passing through the central column into the thermal range so as to enhance the possibility of producing a fission.

The central column is 19 in. long and is constructed out of high-purity boron-free graphite available in 4-in.-square by 48-in.-long logs. These same graphite logs are used in the construction of the main graphite pack and of the external thermal column.

Guide Tube and Sample Loader. A 4-in.-diameter aluminum tube passing through the center of the central thermal column serves to form the horizontal passageway for sample loading into the core. This tube, called the loading tube, extends from the control face of the reactor through the main graphite assembly to the front face of the main graphite pack (Fig. 4-9).

It is apparent that the horizontal arrangement readily lends itself to adaptation to production-line testing of many forms of fissionable material.

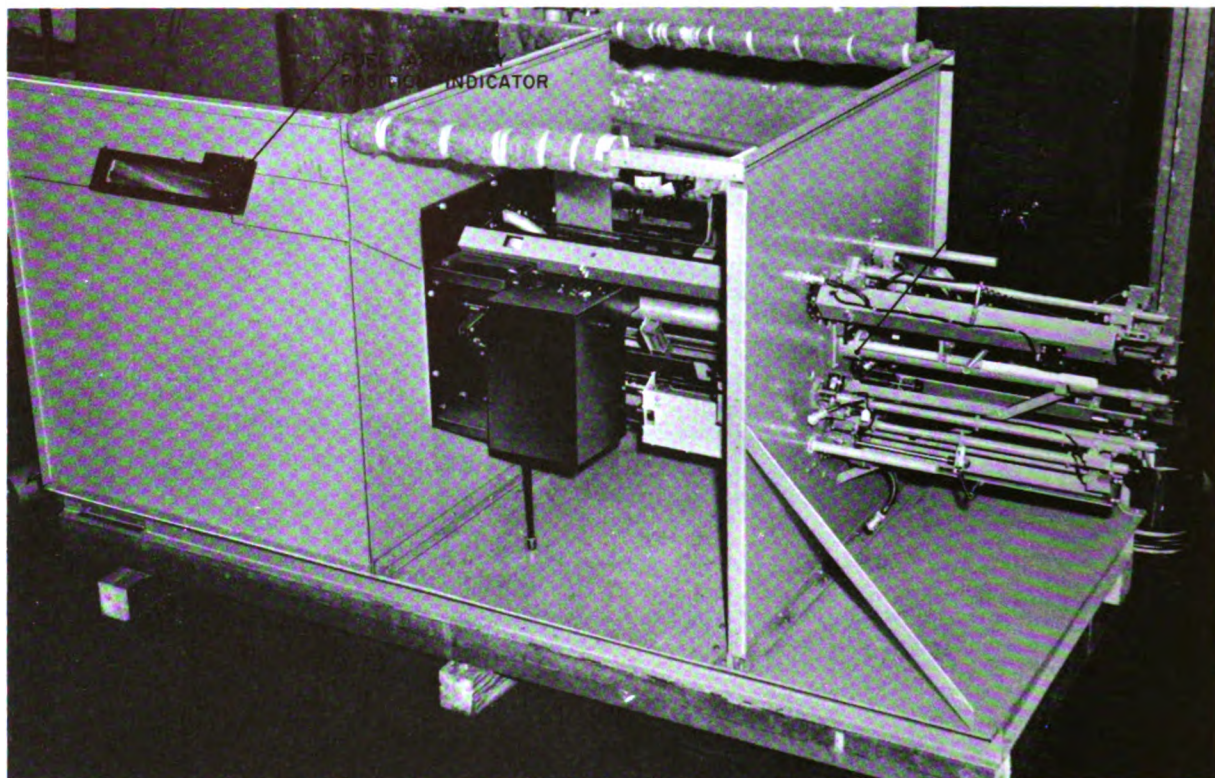


Fig. 4-9 Nuclear test reactor.

The sample loader (illustrated in Fig. 4-10) consists of a 42-in.-long, $3\frac{3}{4}$ -in.-diameter graphite rod which is used as the carrier for inserting and withdrawing samples out of the active area. It consists of three short rods dovetailed together to obtain flexibility in locating the sample holder

It functions to reduce further to thermal energies the percentage of fast neutrons that have managed to escape through the main pack. A hole of the same diameter as the main graphite-pack feed hole is machined through the thermal column and is aligned with the main hole. This column

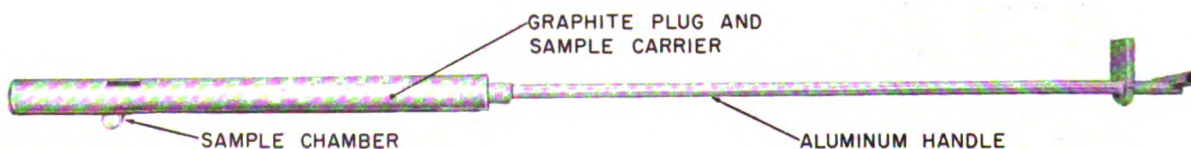


Fig. 4-10 Sample-loading tool.

anywhere within the core. During operation, unused portions of the guide tube are filled by the short graphite rods.

External Thermal Column. A 4-ft cube of high-purity graphite, called the external thermal column and shown in Fig. 4-1, is located against the front side of the main graphite moderator.

is constructed of the same material as the main moderator pack and is stacked in a similar manner.

A 12-in. rectangular opening traversing the length of the thermal column and terminating at the reactor face is provided for large sample irradiation. The column is constructed in a manner that permits the use of this large opening without complete disassembly of the thermal column.

INSTRUMENTATION

The electronic instrumentation provided for use in the NTR is adapted for monitoring startup, steady-state power-level operation, and change of power level. The instrumentation also includes the safety circuits required for the protection of the reactor and its operating personnel. It consists of the following assemblies:

1. Counting-circuit assembly.
2. Power-level trip assembly.
3. Log N and period trip assembly.
4. Difference circuit.
5. Audible indicator.

Counting-circuit Assembly. The counting-circuit assembly shown in block form in Fig. 4-11 consists of a fission chamber, suitable pulse preamplifiers and linear amplifiers, a scaler, and a high-voltage power supply.

At the time of reactor startup, the neutron-flux density in the reactor is largely dependent upon the value of the neutron source in the reactor. Suitable sensitive equipment is therefore necessary in order to detect this low order of activity.

The counting circuits provide a method of monitoring this subcritical condition by amplifying the feeble ion-chamber pulses and applying them to a scaler and audible indicator.

The audible-indicator circuit converts the counting pulses into sound pulses of variable pitch. Immediate information on the subcritical condition of the reactor is obtained by noting the change in the pitch of the sound pulses.

During startup the use of a multiplicity of counting channels is required to determine accurately the critical mass of the virgin reactor. The number of counting circuits used at startup may be in the order of 4 or 5. After the determination of the critical mass, one counter will suffice.

The components in the counting circuit are:

1. An enriched U^{235} fission chamber.
2. A preamplifier.
3. A linear amplifier.
4. A 10^4 scaler.
5. An audible indicator.

Fission chamber. The fission chamber is a hermetically sealed all-aluminum unit containing

approximately 2 mg/cm² of U^{238} enriched in the U^{235} isotope. The filling gas of the chamber is 99.8 per cent pure nitrogen, and it possesses a sensitivity of 1 count per unit neutron flux. The detector can cover a range to 10^5 cps.

Preamplifier. The preamplifier is an impedance transformer which isolates the pulse signals generated in the neutron chamber and applies them to the linear amplifier. The preamplifier will provide a usable signal for the linear amplifier when it is coupled with approximately 80 ft of coaxial cable.

Linear amplifier. The linear-amplifier assembly consists of two parts: the linear amplifier proper and a discriminator. The function of these two components is to provide further amplification of the applied pulses and to discriminate against pulses of various voltage levels. Since undesirable gamma background signals will always be generated in the chamber and associated circuits, the discriminator is adjusted to suppress these unwanted signals and to pass on to the scaler the proper limited signal. The discriminator also serves to clip the incoming pulses. The amplifier gain and the discriminator safety levels are both variables which result in a wide range of input sensitivities through a combination of gain and level settings.

In the preamplifier, the input resistor and the total cable capacitance of the input circuit determine the width of the signal pulse. The pulse should not be greater than the resolving time of the scaler that is used. In this application, the signal pulse obtained was essentially 2 to 3 μ sec wide, whereas the resolving time of the scaler was about 6 μ sec; hence this criterion is satisfied.

The preamplifier used with the counting circuit had a bandwidth of approximately 1 megacycle.

Scalers. The scaler receives its input signals from the discriminator described above. Provision is made in the scaler for extracting a signal to drive the audible indicator. Pulses for the audible indicator are taken from the scaler at the 1, 10, 100, and 1000 counting points. These output points increase the audible-indicator range. A panel selector switch determines which of these pulse inputs is used.

Power-level Trip Assembly. The power-level trip assembly is illustrated in block form in Fig.

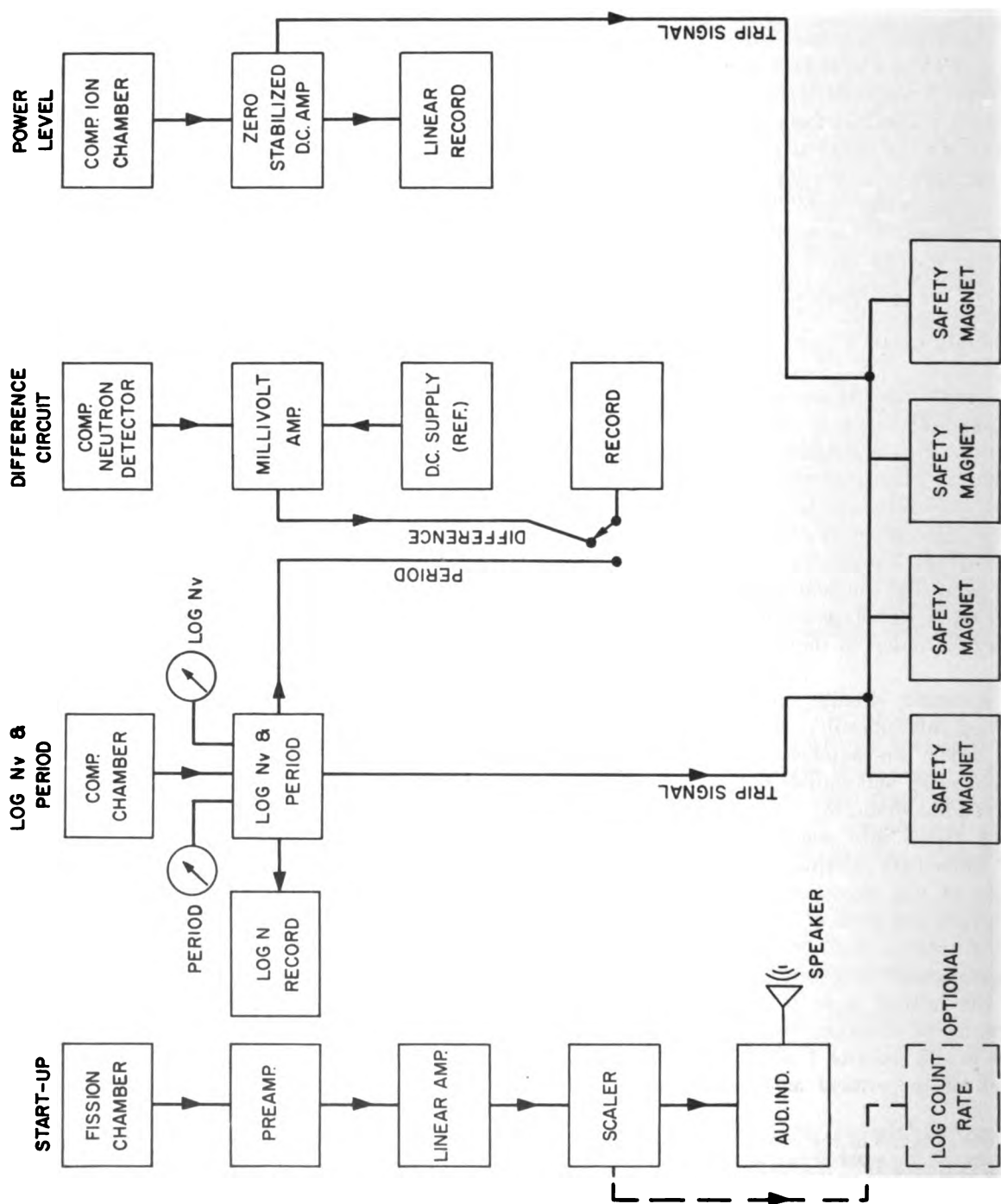


Fig. 4-11 Reactor control system.

4-11. It consists of a compensated neutron detector mechanically adjusted to allow a compensation better than 90 per cent, suitable amplifying equipment, and high-voltage supplies. The power-level trip circuits provide an accurate measurement of reactor activity at detectable power levels. This assembly also provides a "scram" signal if the power level exceeds some predetermined level. A recorder attached to the indicator provides a permanent record of the operating power level. The level trip assembly consists of the following components:

1. A compensated ionization chamber.
2. A zero-stabilized d-c amplifier.
3. A power-level trip circuit and power supply.
4. A recorder.

The compensated ionization chamber is a hermetically sealed nitrogen-filled detector constructed completely of aluminum. It is mechanically compensated and has a sensitivity of 3×10^{-14} amp per unit neutron flux. The zero-stabilized d-c amplifier is essentially an accurate and fast-responding electrometer amplifier coupled to a narrow-band zero-correcting a-c amplifier. Plug-in output trip circuits are available with this amplifier to provide linear output signals, relay signals, or electrical step-function signals.

Log N and Period-trip Assembly. This equipment measures and records the logarithm of the current from the d-c ionization chamber and also presents the information on an instrument. The assembly covers a current range of 6 decades and hence can handle 6 decades of reactor power. It also measures and records the time rate at which the power level is changing. If this period is smaller than some preset value, the reactor will scram. The period of a reactor is designated as that time required for the exponentially rising power to change by a factor of 2.718. A panel instrument indicates the reactor period in seconds, and another panel meter indicates the logarithm of the power level.

The period-trip assembly consists of a compensated ion chamber, a logarithmic amplifier and period-trip circuit, a recorder, and a highly regulated power supply. The logarithmic amplifier and the period-trip chassis consist of three parts: (1) a logarithm amplifier; (2) a differentiating network; and (3) a vacuum-tube voltmeter with a

relay. The logarithm amplifier covers a current range from 10^{-11} to 10^{-4} amp and has its output metered and recorded. The differentiating network which follows the logarithm circuit produces an output voltage which is the differential of its input. This output voltage is measured by the vacuum-tube voltmeter to indicate the reactor period and is also recorded. If the differentiated voltage exceeds a predetermined value, the tube relay drops out and causes the reactor to shut down.

The logarithmic characteristics of the amplifiers are obtained by taking advantage of the characteristics of the vacuum-tube diodes when they are biased in reverse. When reverse bias is applied to a suitable diode, the output voltage across the diode is proportional to the logarithm of the input current over a wide range of input currents.

Difference Circuit. In order to maintain manually the power level of the reactor at some constant value, a difference circuit is incorporated. This particular assembly indicates to the operator the fluctuations of the reactor-flux level above and below some established reference value. A reference battery current flowing in opposition to the generated chamber current produces a differential value of current which is amplified, recorded, and indicated on a panel instrument. Any change in the reactor power level will show up as a change in this differential current. Zero differential current over a period of time is an indication that the operating power level is constant. The differential circuit can be easily adapted to an automatic-flux servo system. The components of the difference circuit are:

1. A compensated-neutron detector.
2. A highly regulated high-voltage supply.
3. A d-c millivolt meter and amplifier.
4. A regulated chamber high-voltage supply.

The d-c source consists of a battery and a resistor assembly mounted integrally in a copper-lined box and connected to provide a variable range of reference current. The d-c millivolt meter and amplifier are standard commercial units. The high voltage for the d-c neutron detector can be supplied by line-connected units or by batteries. Battery operation has the advantage that the noise generated in the circuit is lower. The batteries can be located directly at

the neutron detector, thereby reducing cable length and also reducing another possible source of noise. Since the current consumption of the average chamber is extremely minute, the life of these batteries can be considered equivalent to their shelf life.

Audible Indicator. The audible indicator is designed to provide an audible signal whose frequency is an indication of reactor activity. The response of the human ear is such that as the sound increases in pitch while its amplitude remains constant, the apparent loudness will seem to increase. The audible indicator is designed to

eliminate this effect by reducing the gain of the circuit as the frequency of the input signal is increased. The gain at the high-frequency end is adjusted by a HI-RATE control and the gain at the low-frequency end is controlled by a LO-RATE control. The audible indicator consists of the indicator circuit, an audio amplifier, and a speaker.

The electronic instrumentation outlined above is contained in two sets of relay racks located in the control room. Figure 4-12 illustrates a typical control-console assembly minus the startup equipment, which for this particular installation was located elsewhere.

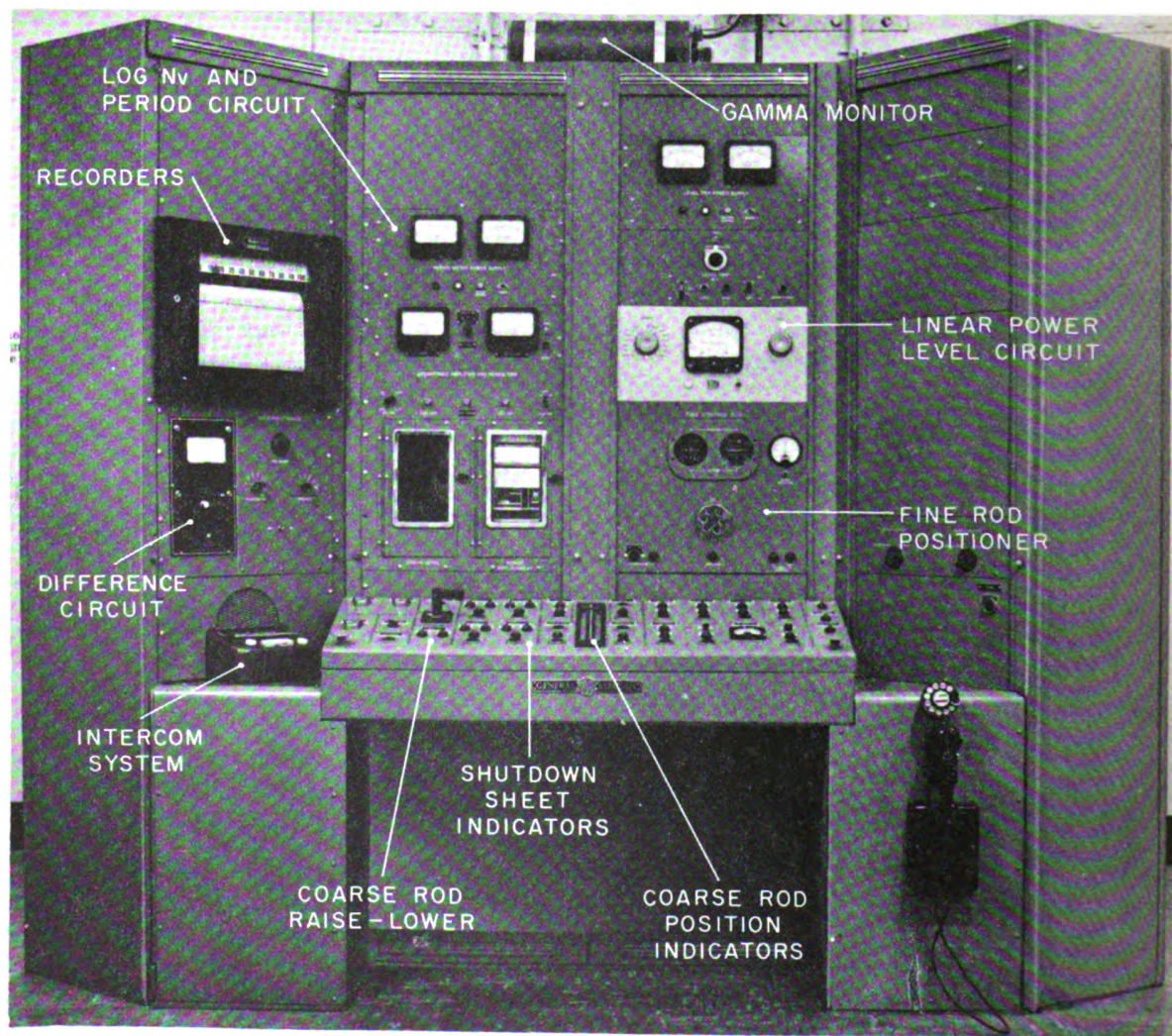


Fig. 4-12 Control console.

REACTOR CONTROL

The safety and control functions of this reactor are entirely independent except that all safety mechanisms must be in the scram position before the control system can be operated. The startup procedure is as follows:

1. Before the reactor can be operated, the fine rod, coarse rods, safety rods, source rod, and shutdown sheets must be completely in the reactor.

2. The fine-rod and coarse-rod mechanisms can next be tested while all safety rods and shutdown sheets are in.

3. After testing, the control rods are returned to the IN position and the four safety rods are brought out separately and cocked. This sets up the scram circuit of the reactor, which is indicated by a PILE READY light on the control desk, provided all reactor-room doors and other safety interlocks are closed.

4. The six shutdown sheets can now be run out in sequence. When all sheets are out, the source rod and coarse-control rods can be taken out. The coarse rods move out in precision 1-in. increments. The fine rod may be energized when the coarse rods are on any of the 1-in. stops.

Shutdown-sheet Mechanism. Figure 4-6 shows the peripheral arrangement about the core of the six shutdown sheets. The shutdown sheet is attached to a lead screw which is driven by a belt and pulley coupling to a $\frac{1}{12}$ -hp electric motor. The lead-screw bearing is securely fastened to the reactor frame. Each shutdown sheet is worth approximately $0.01 \Delta k/k$. The sheets slide in graphite ways machined in the reflector pack. The sheets and drives require a minimum of maintenance and test and no calibration.

Figure 4-7 shows a shutdown sheet in its milled slot outside the core cavity. Figure 4-13 shows the six shutdown-sheet drive mechanisms mounted on the reactor frame. Figure 4-14 shows lead-screw housings and limit switches extending outward from the front face of the reactor frame.

Safety-rod Mechanisms. There are four safety rods in this reactor situated peripherally about the core cavity. Figure 4-15 shows a safety-rod mechanism. The rods themselves are $\frac{1}{2}$ -in.-diameter stainless-steel-clad cadmium rods mov-

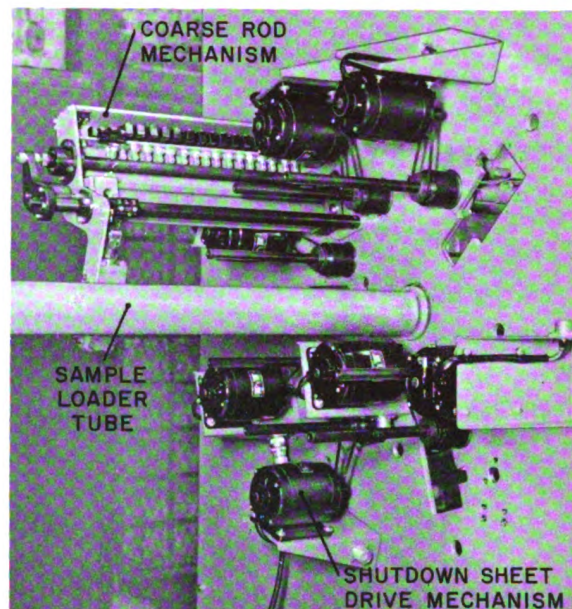


Fig. 4-13 Loading panel, rear view.

ing in aluminum guide tubes. The rods are picked up and held by an electromagnet, which in turn is driven out by a pneumatic cylinder. The retraction motion stores energy in two clock springs, either of which will supply sufficient force to drive the rod into the core. The rods are held in a cocked position by the electromagnets until a scram signal deenergizes the magnet. The pneumatic cylinders are controlled by push buttons on the desk. Deceleration of the rods is accomplished

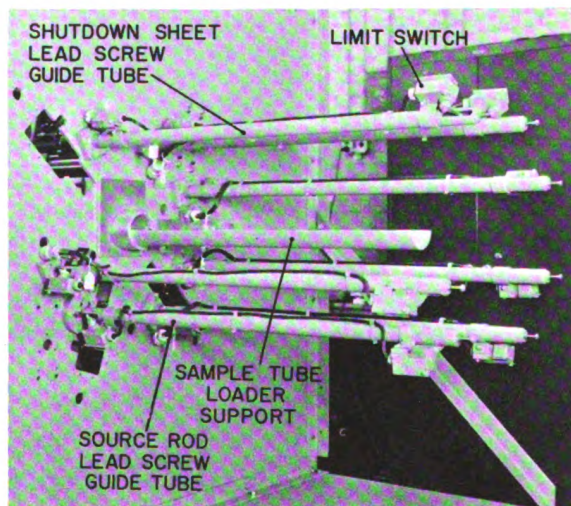


Fig. 4-14 Loading-panel control mechanism.

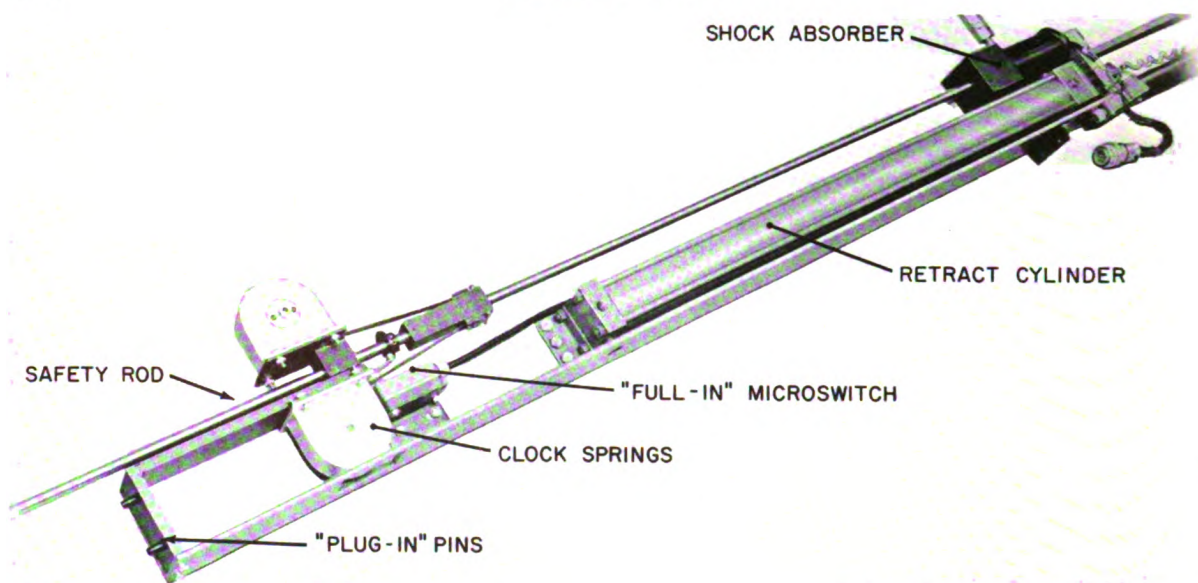


Fig. 4-15 Safety-rod assembly.

by an air dashpot. As shown in Fig. 4-16, the total time for rod insertion, including instrument response time, is 300 msec. The four rods will introduce a total negative fractional reactivity of 0.015 and any two of the rods are sufficient to shut down the reactor. Here again the design includes simplicity and reliability. Models of the mechanism have been tested for 25,000 cycles and failed to exhibit appreciable wear or operating difficulties.

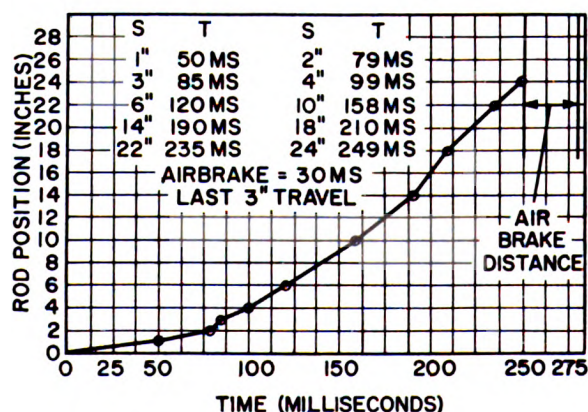


Fig. 4-16 Safety-rod speeds.

Coarse- and Fine-rod Mechanism. The reactor control system has the high precision necessary to provide the reactivity sensitivity required

for analytical work in fuel quality control and nuclear cross-section determinations. In addition, the control system must possess large reactivity. The control-rod system possesses a control reactivity of $0.0155 \Delta k/k$ and remotely indicates the position to within an error of 1 part in 150,000. This accuracy is obtained by positioning the coarse-control rods absolutely. The coarse rods are approximately 16 in. long and can take up positions only at 16 different intervals along their travel. These intervals are 1 in. and are determined by the position of stopblocks located to within 0.0001 in. The total reactivity worth of the fine rod is about one-sixth the worth of a coarse rod. The fine-rod reactivity covers the 1-in. intervals of coarse-rod reactivity and is positioned to 0.001 in.

The coarse-rod mechanism is shown in Fig. 4-17. The coarse rods are cadmium-filled stainless-steel tubes. The extension of the rod has several Carboloy pins spaced at 1-in. intervals. The coarse rod is moved out by the pneumatic cylinder, which is controlled by push buttons on the desk. The cylinder forces the rod out until one of the pins strikes the stopblock. At this time a contact switch lights up the ON STOP light. To move the rod to the next stop, the ROTATE cylinder rotates the rod until it slides past the stop and moves to the next pin position. The

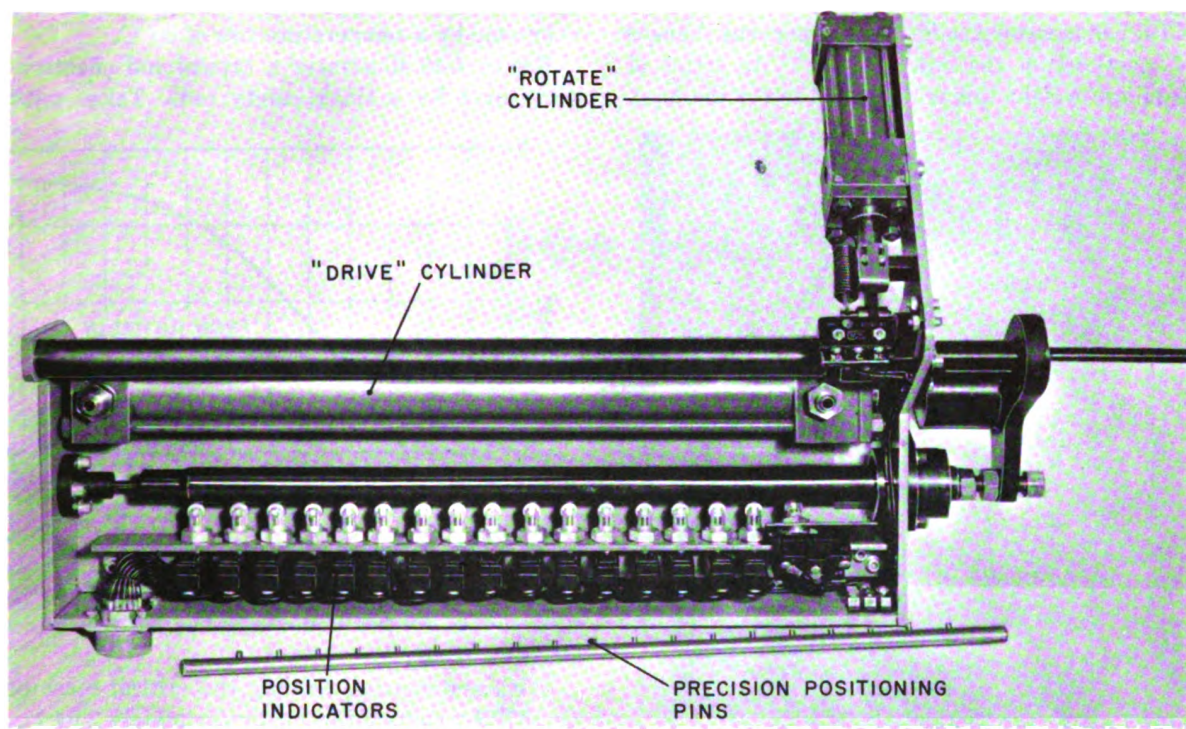


Fig. 4-17 Coarse-rod actuator.

limit switches indicate which stop the rod is on. The rod may be moved to the IN position with one continuous motion of cylinder. The pneumatic speed-control valve permits a variation in rod withdrawal speeds. The IN speed is not controllable and requires about 9 sec for complete insertion. The negative reactivity of the coarse rod is not depended upon to scram the reactor. The two strips of indicator lights located on the center of the desk shown in Fig. 4-12 are the position lights for the two coarse rods.

Source-rod Mechanism. The function of this mechanism is to ensure the presence of a neutron supply during startup operations and to remove the source after startup, when its neutrons could interfere with low-power experiments. During startup the source provides the neutron monitors with a detectable flux when the reactor is completely shut down. The source-rod mechanism is identical to the shutdown-sheet mechanism except for its mounting plate and rod.

In Fig. 4-14 the source-rod mechanism can be

distinguished from the shutdown-sheet mechanisms by its round mounting plate.

COOLING SYSTEM

The cooling system of the reactor shown in Fig. 4-18 is simple, requiring a laminar flow of water at about 12,000 lb/hr, with an inlet temperature of 90°F and an outlet temperature of 104°F. A gear pump circulates deionized water through the core. Heat is removed in a water-to-water heat exchanger of about 30-ft² area. The water inlet and outlet pipes can be seen in Fig. 4-7. The pressure drop across the core is about 0.005 psi.

REACTOR PERFORMANCE

A change in reactivity, $\Delta k/k$, in the NTR of 8×10^{-3} is produced when an impurity having a cross section of 16 cm² is introduced into the central region of the core. The minimum reactivity change that can be detected by the NTR is 10^{-7} , when 0.0016 cm² of absorber cross section is in-

roduced into the central thermal column. The NTR will measure accurately to 1 per cent changes of reactivity of the order of 10^{-5} . In terms of cadmium weight this reactor will detect the pres-

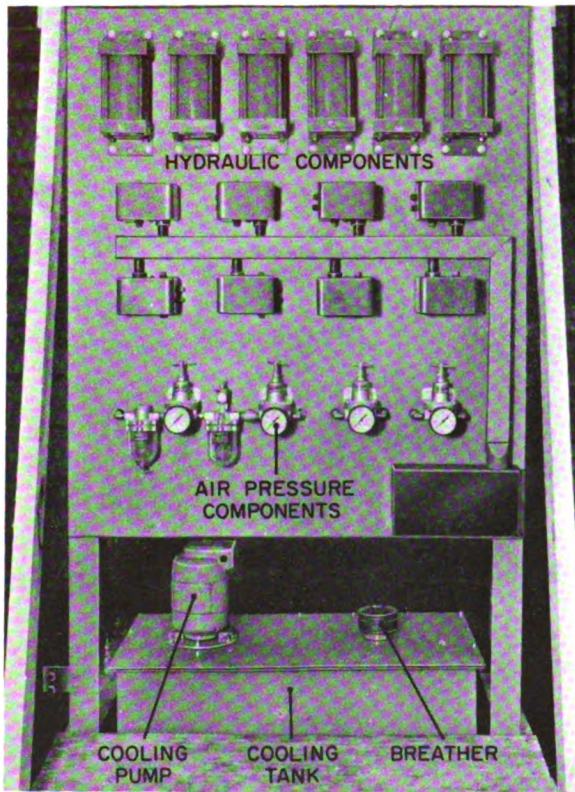


Fig. 4-18 Cooling system.

ence of $120 \mu\text{g}$ of cadmium placed in the central column and will measure to within 1 per cent changes in reactivity produced by 12 mg of cadmium. This value decreases per square centimeter for larger samples and increases for thinner samples. An average routine test requires 30 min to obtain a detection of a $\Delta k/k$ of 10^{-7} . A change of reactivity in the order of 0.0015 to 0.002 is a reasonable but maximum upper limit established for routine measurements since the period generated by this change of reactivity can be easily controlled by the operator.

An estimate of the reactivity change due to the temperature produced when oil is used in the active lattice may be obtained from

$$\frac{\Delta k}{k_{\text{eff}}} = -10^{-4} \Delta T(^{\circ}\text{C})$$

A built-in excess k of 0.006 can be completely overcome by a temperature rise of 60°C .

Figure 4-19 illustrates a typical rod effectiveness curve for a single safety rod. Taken with

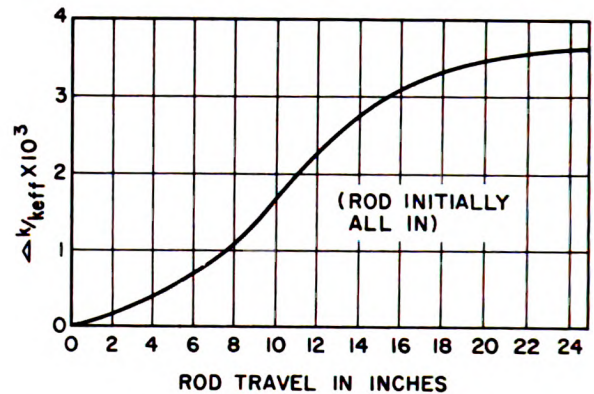


Fig. 4-19 Safety-rod position vs. reactivity change.

Fig. 4-16, it reveals that approximately 200 msec after scram 0.8 per cent $\Delta k/k$ has been inserted into the core.

With symmetrical loading the critical mass of the NTR was found to be approximately 2.54 kg. With this loading the k_{eff} of the core was 1.004.

Near criticality, one fuel disk (5.7 g) is worth approximately 0.076 per cent Δk .

The neutron source was a polonium-beryllium mixture producing approximately 10^6 neutrons/cm²/sec.

The neutron flux distribution in the NTR on a per-watt basis is as follows:

1. Center of core:

$$\varphi_{\text{total}} = 3.7 \times 10^7 \text{ n/cm}^2/\text{sec}$$

$$\varphi_{\text{fast}} = 0.75 \times 10^7 \text{ n/cm}^2/\text{sec}$$

$$\varphi_{\text{thermal}} = 3 \times 10^7 \text{ n/cm}^2/\text{sec}$$

2. Fuel lattice:

$$\varphi_{\text{total}} = 1.2 \times 10^7 \text{ n/cm}^2/\text{sec}$$

3. At face of graphite moderator:

$$\varphi_{\text{total}} = 3 \times 10^6 \text{ n/cm}^2/\text{sec}$$

$$\varphi_{\text{fast}} = 3 \times 10^4 \text{ n/cm}^2/\text{sec}$$

4. At end of the thermal column:

$$\varphi_{\text{thermal}} = 10^5 \text{ n/cm}^2/\text{sec}$$

APPENDIX

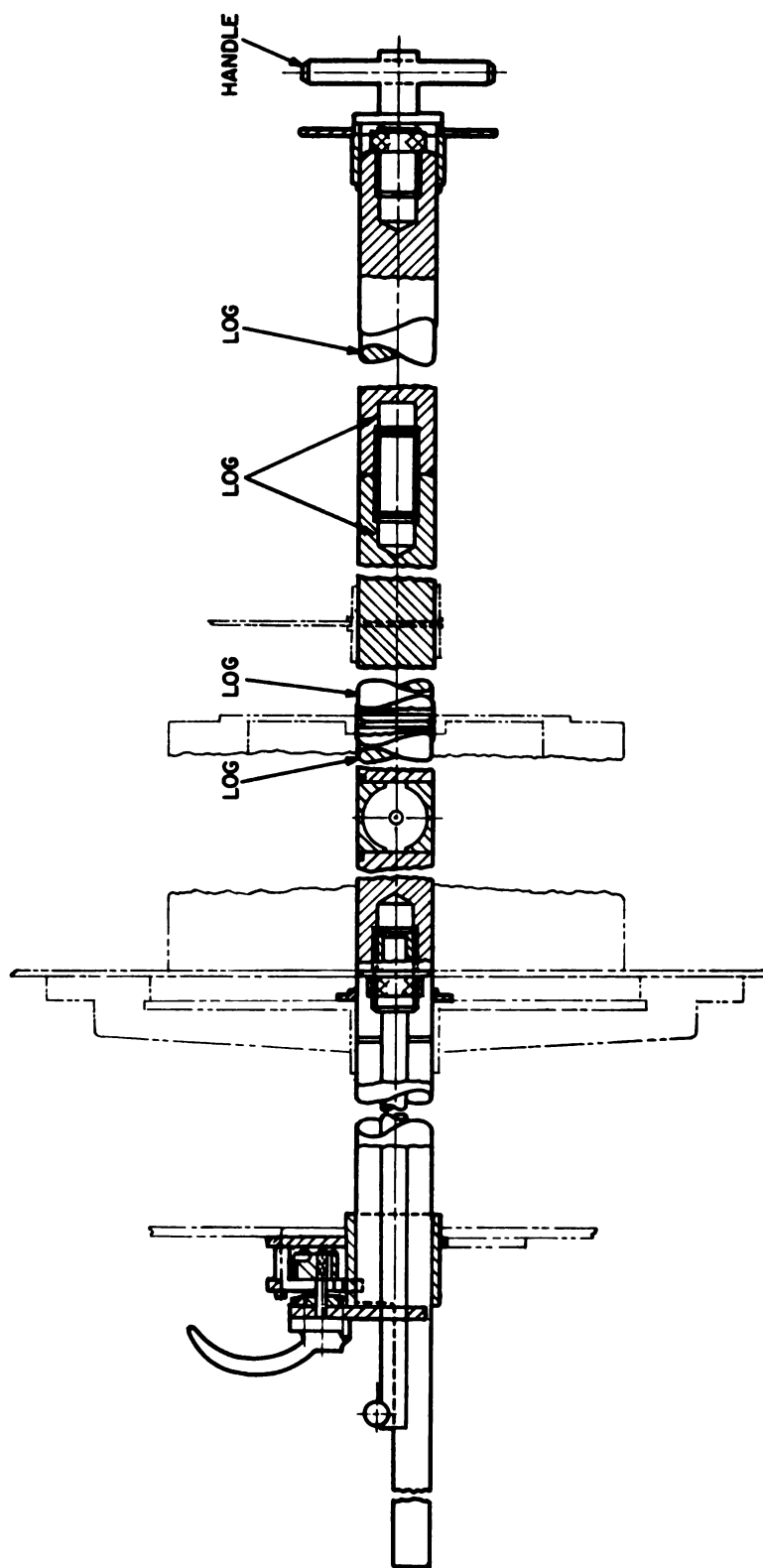


Fig. 4-20 Sample tube loader.

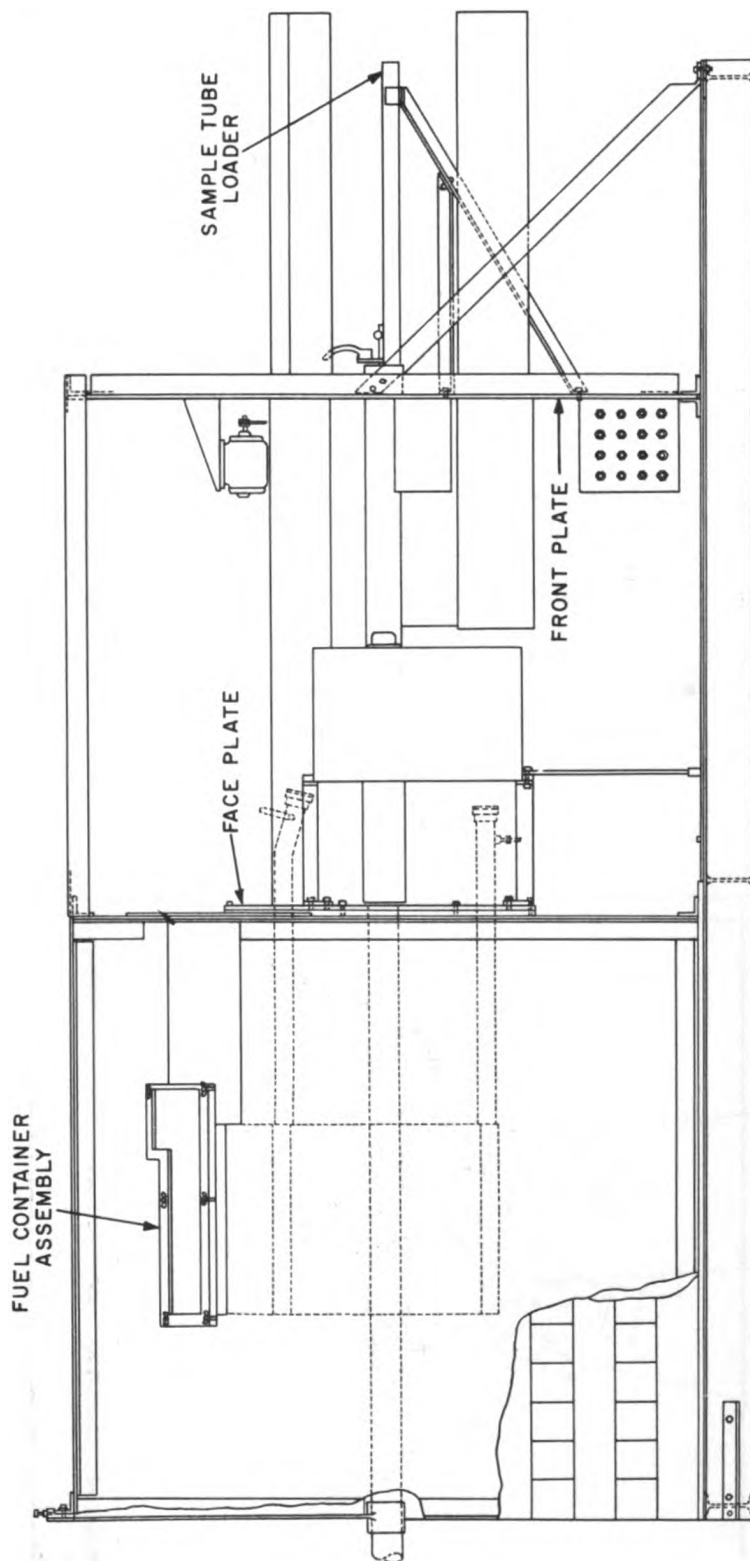


Fig. 4-21 Reactor, side view.

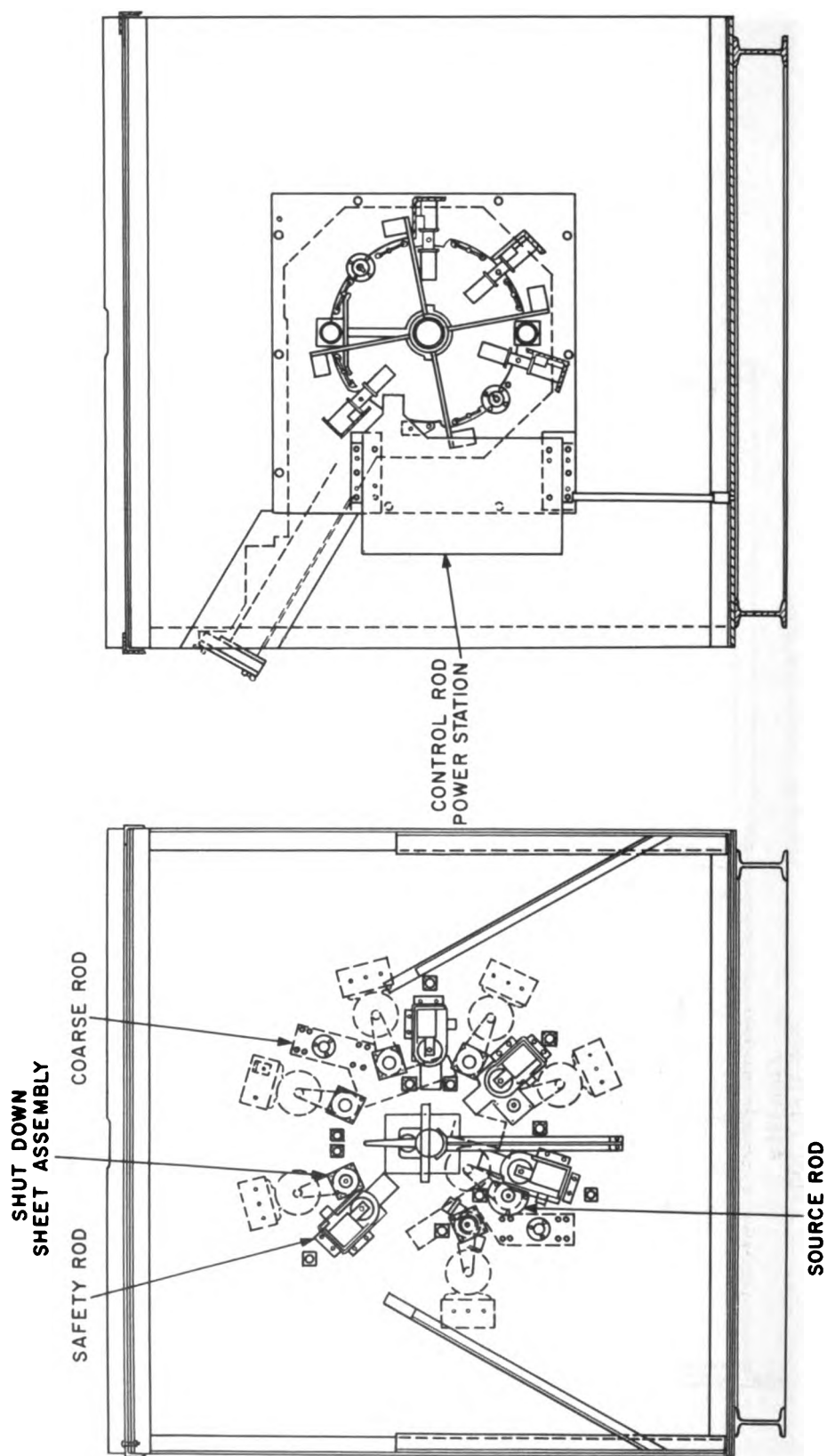


Fig. 4-22 Reactor, end view

CHAPTER 5

Heavy-water-moderated Reactor

Heterogeneous—Enriched Fuel

**COMPILED BY ARGONNE NATIONAL LABORATORY
UNIVERSITY OF CHICAGO**

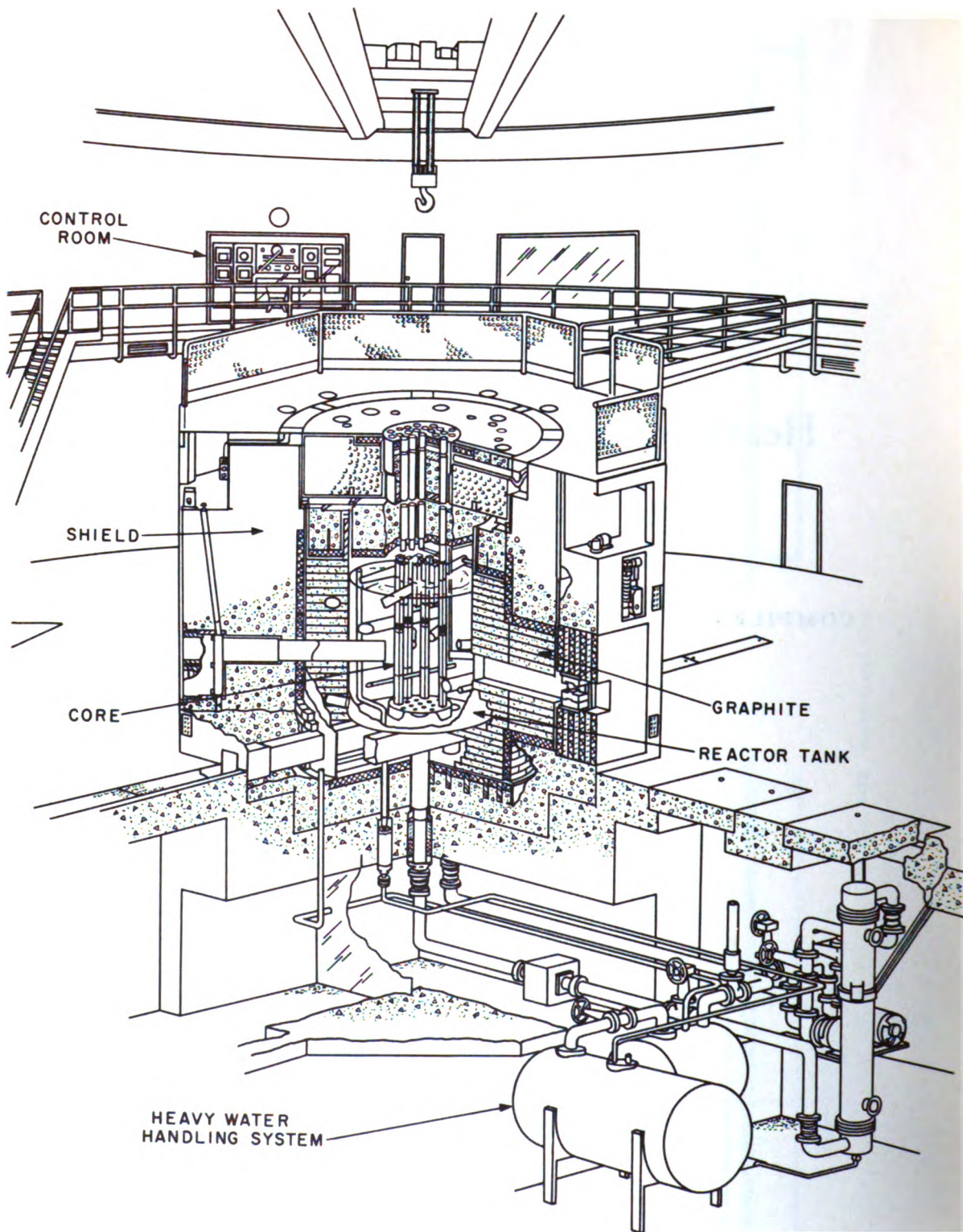


Fig. 5-1 Heavy-water-moderated reactor: heterogeneous—enriched fuel, isometric.

CHAPTER 5

Heavy-water-moderated Reactor

Heterogeneous—Enriched Fuel

The Argonne National Laboratory research reactor (CP-5), shown in Figs. 5-1 and 5-2, was designed and built by the Argonne National Laboratory. This reactor has been operated by the ANL since February, 1954, as a research facility in a

broad program of neutron physics and reactor development requiring a high neutron intensity at a reasonable power level.

The CP-5 is a heavy-water-moderated reactor fueled with enriched uranium alloy and cooled by

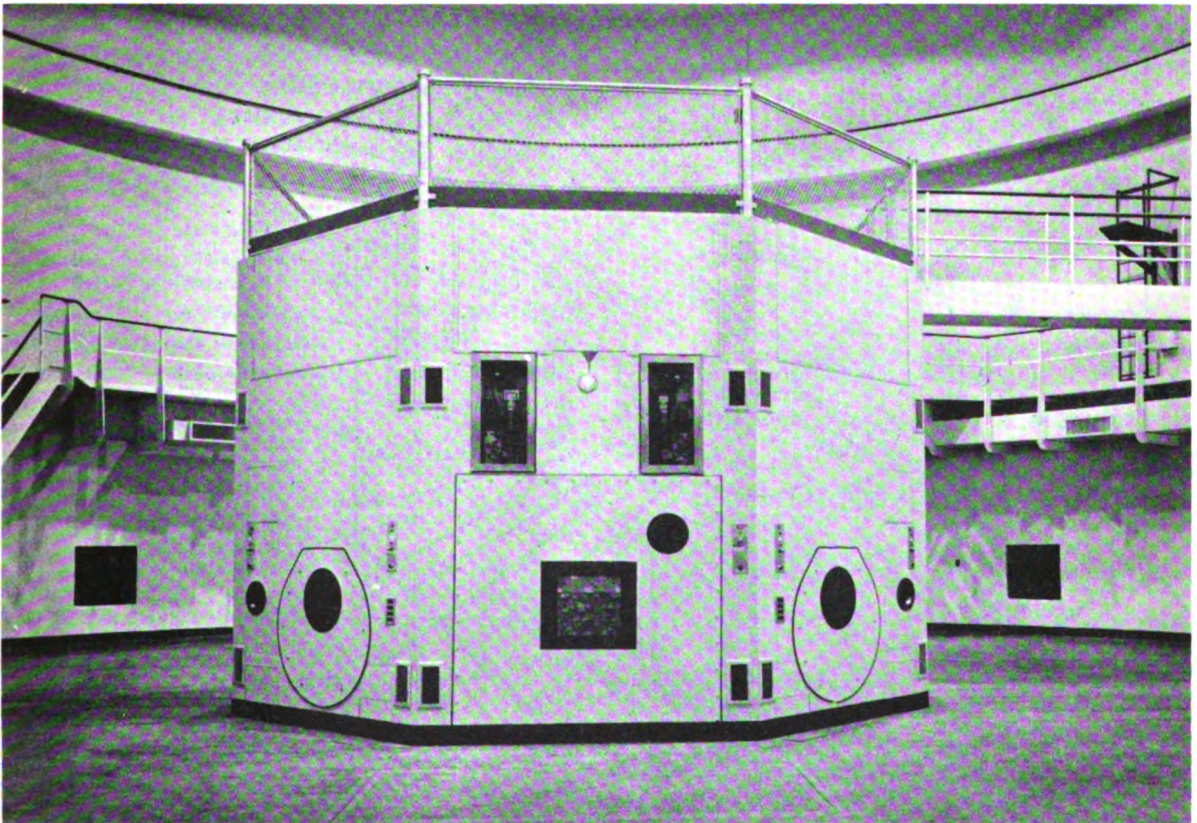


Fig. 5-2 Completed reactor.

heavy water. The reactor has a maximum thermal-neutron intensity of 2×10^{13} neutrons/cm²/sec at the thimbles just external to the fuel zone, when operating at a 1000-kw heat-removal rate. The reactor is so dependent on the presence of the heavy-water moderator that it will shut itself down by steam formation even under the most drastic alterations in chain-reaction behavior.

Experimental facilities for sample irradiation or beam work are provided by 30 vertical thimbles, 4 horizontal beam holes, 4 horizontal test holes, 2 horizontal pneumatic tubes, and 2 thermal columns.

GENERAL DESCRIPTION

In the CP-5 the U²³⁵ is dispersed in aluminum plates mounted so that the core approximates a

cylinder 2 ft in diameter and 2 ft high. The fuel-bearing plates are mounted in several rectangular tubes, or boxes, of thin aluminum plate. Each box is individually replaceable, thus permitting changes in the amount of uranium in the reactor to be made in small steps. For initial operation there are 10 fuel plates per box. Twelve boxes are expected to provide enough fissionable material for useful operation of the reactor.

Each box is part of a fuel assembly. Aluminum extensions on the box and a top cylindrical piece containing shielding materials complete the unit. Each assembly is held in a vertical position by confinement of the top cylinder and by contact between a bottom adapter and the coolant-distribution chamber. In the cover of this chamber there are 17 holes for fuel-assembly location. The pattern of assemblies results in five parallel rows

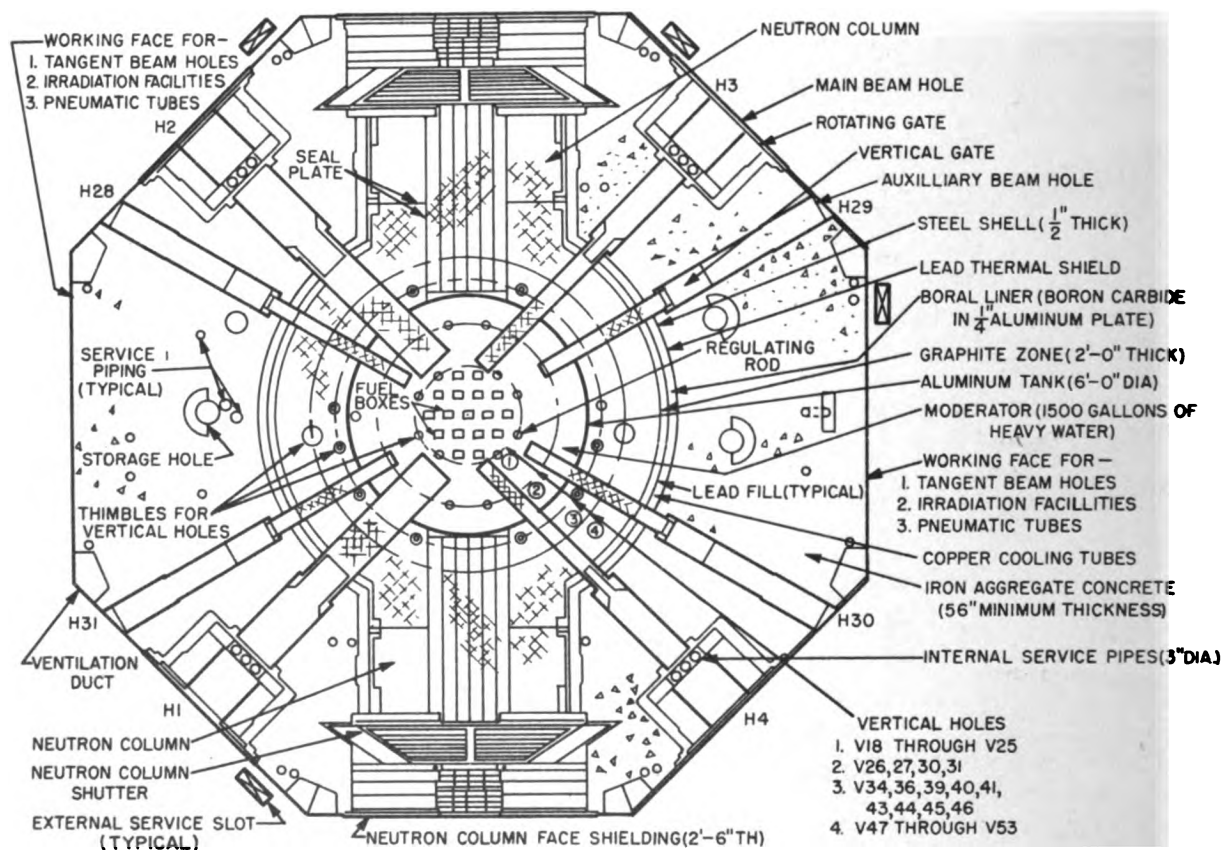


Fig. 5-3 Horizontal section through reactor.

in one direction, as may be seen at the center of Fig. 5-3.

Four control rods of the signal-arm type operate between the parallel rows of fuel assemblies. These are strong neutron-absorbing units known as shim-safety rods. A fifth unit for fine control, called a regulating rod, moves up and down a short distance outside the fuel-assembly array.

The core of the reactor is at the center of a 6-ft-diameter aluminum tank. Heavy water, acting as coolant, flows upward in each fuel assembly before discharging into the tank. About 15 per cent of the moderator in the core is thus confined near the heat-generation surfaces. The heavy water flows out through a pipe at the bottom of the tank to deliver the heat generated to a light-water circuit and then returns to the plenum chamber. The level in the tank is maintained at a fixed height of $6\frac{1}{2}$ ft during operation.

The region of heavy water above the core is occupied by the shim-safety rods but the region 2 ft thick around and below the core is available for experimental use. Numerous tubes penetrate this zone for isolating small experimental volumes from the bulk of the heavy water. The region of useful neutron intensities is augmented by a graphite zone 2 ft thick extending around and below the tank of heavy water. This zone is 10 ft high and 10 ft in diameter.

Outside the graphite, the neutrons and gamma rays are absorbed in a massive shield. The thermal-neutron intensity is first reduced about 100,000-fold by boron carbide embedded in $\frac{1}{4}$ -in.-thick aluminum. The gamma rays are next reduced below the point of significant heat generation by $3\frac{1}{2}$ -in.-thick lead. The rest of the shielding is a special concrete 56 in. thick, with iron punchings and hydrated iron ore as aggregate for slowing down the significant portion of the residual fast neutrons from the core and for absorbing gamma rays and thermal neutrons. The total shield thickness is thus about 5 ft. Externally, the reactor is an octagonal prism 20 ft wide and $13\frac{1}{2}$ ft high, as is shown in Fig. 5-2.

The Fuel Assembly. The various parts of the fuel assembly are delineated in Fig. 5-4. The cylindrical shield piece at the top is called a weighting plug. Exerting a load of 115 lb on the seal surfaces at the bottom of the assembly, it

retains the seal against the upward thrust of the heavy-water coolant. This seal between the fuel assembly and the distribution, or plenum, chamber is made with pipe-union-type mating surfaces.

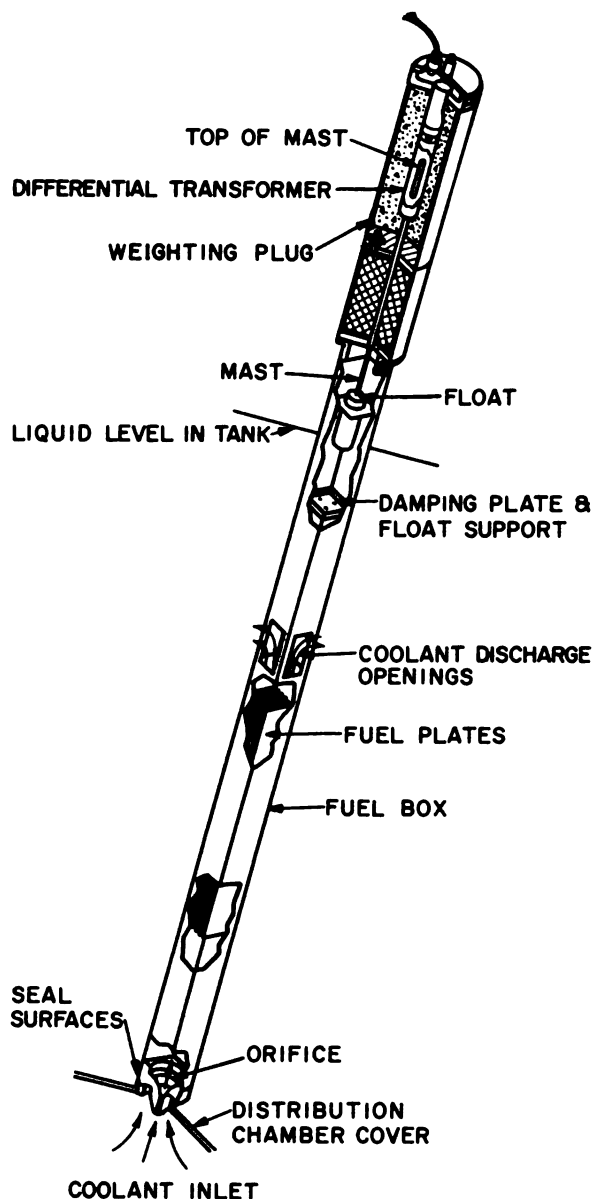


Fig. 5-4 Fuel assembly.

The coolant entering an assembly first passes an orifice for balancing flow, then cools the active plates, and finally discharges through ports into the main body of heavy water. The hydraulic force of the upward flow causes the water level

in the assemblies to be several inches higher than the level in the tank. This added height is measured by a float with a stainless-steel mast entering a transformer in the weighting plug. The response of the transformer is modified to give a linear indication of flow rate.

A spent-fuel assembly is expected to have about 20 per cent of the original U^{235} converted into fission products; a 7-ton portable lead shield is required to transport it to storage. At the anticipated burn-up value, the unloading and refueling operation is performed at about 8-month intervals, on the basis of a 1000-kw power level.

Control Rods and Related Reactor Features.

Reactor control, understood quantitatively in terms of reactivity, is a measure of the ability of a reactor to perpetuate the fission chain reaction. A nuclear reactor with a reactivity of unity is maintaining the reaction at constant power. The reactivity must be greater than unity to increase power level, less than unity to decrease power level. The necessary variation from unity to alter power level conveniently is less than 0.5 per cent. However, under certain conditions, there is excess reactivity of several per cent in the CP-5 reactor, which is held in check by the neutron absorption of control rods. This excess is created by adding more uranium than is barely required for unity reactivity in order to compensate for several losses, including (1) depletion of the uranium, (2) moderator-temperature increase to operating level, and (3) growth of fission-product neutron absorbers, mainly xenon.

The control features available to the operator for altering the reactivity of the reactor are (1) four shim-safety rods, (2) one regulating rod, (3) a quick-opening valve and limited-capacity storage tank for rapidly draining the heavy water from the aluminum tank down to the level of the top of the fuel plates, and (4) a chilled-water system for limiting the rate at which the moderator temperature rises to the operating level to compensate for reactivity gained during startup under special conditions of fission-product effect on reactivity. Each of these control features will be discussed in more detail.

The purposes of the shim-safety rods (Fig. 5-5) are twofold: (1) to adjust for gross changes in reactivity and (2) to shut the reactor down

quickly. To be more specific, the shim-safety rods move away from the fuel to allow the reactivity to remain at unity as the heavy water heats up, the fission products accumulate, and the fuel is burned up. These rods must shut the reactor down and hold it in check even after the water is cooled, the fission products have decayed, and the new fuel has been added. Any two of the four "strong" rods are sufficient to override any accretion of reactivity. All four rods respond as safeties upon a scram or shutdown signal.

The details of the shim-safety-rod system are shown in Fig. 5-5. The control part, or signal arm, contains the strong neutron absorber, cadmium, between sheets of aluminum formed into a flattened tube and sealed to maintain lightness. The upper end of the arm is mounted in ball bearings for ease of rotation. The degree of possible rotation is limited by a bolt entering an arc segment cut out of the upper edge of the rod. Normally, the limits of rotation are determined by the drive mechanism located at the face of the reactor. The signal arm and mechanism are linked by a shaft fitting into the arm in a modified spline engagement. The outer end of the shaft is joined to a ball-nut screw drive. A reversible motor actuates the ball-nut screw through a magnetic clutch. In upward movement the ball-nut screw compresses a spring. In scram action, caused by deenergizing the clutch, the spring assists gravity in moving the signal arm to the center of the reactor. An information train geared to the outer end of the shaft transmits data on rod location to the control room.

The regulating rod is the control element used to make the reactor critical, to keep reactivity equal to unity, and to adjust or alter power level. Since its movement determines the behavior of the reactor, it is faster than a shim-safety rod. However, its reactivity assignment for its full travel of 2 ft is less than the delayed neutron contribution to reactivity. Thus, the rate of rise of power caused by moving the rod to its out position is determined by the relatively slow time constants of the neutrons emitted several seconds after fission.

The neutron-absorbing element of the regulating rod again is a cadmium tube sandwiched between aluminum tubes. The drive and feedback motors and information train are above the reac-

tor. The arrangement permits steady power level to be maintained by automatic control.

The control devices discussed so far depend on absorption of neutrons to alter reactivity. The fact that the height of the moderator can alter reactivity is utilized as an additional safety fea-

tivity is made the basis for reactivity adjustment. The usefulness of the system is limited to compensating for reactivity changes due to certain variations in the amount of the high neutron absorbent, xenon. The loss in reactivity caused by the accumulation of this fission product during a

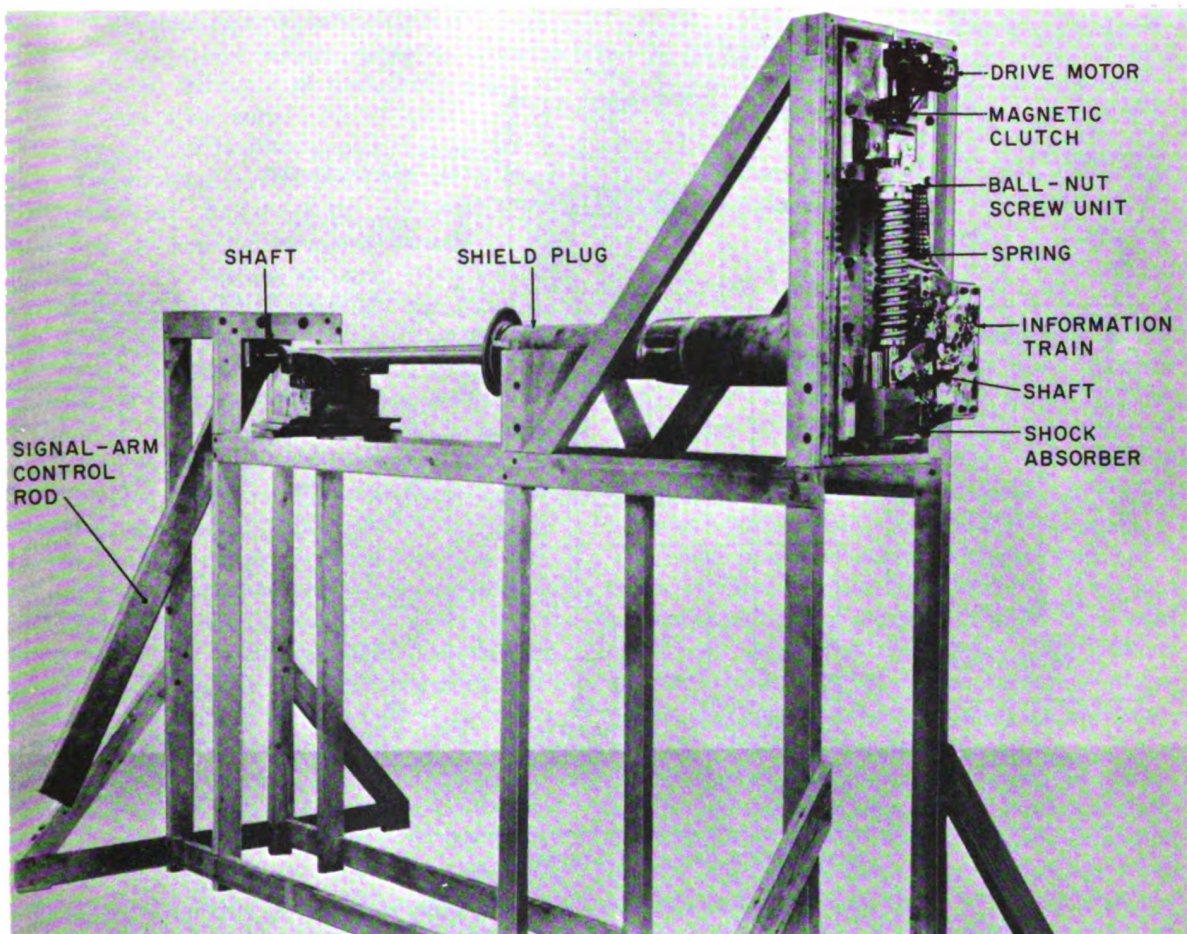


Fig. 5-5 Shim-safety-rod mock-up.

ture. Lowering the level of the heavy water to the top of the fuel plates provides about the same reactivity loss as two shim-safety rods fully inserted. The time necessary for the moderator level to drop the $2\frac{1}{2}$ ft is about 20 sec. An isolated system vented back to the reactor tank prevents any further drop in moderator level that might result in excessive heating of fuel plates.

In another control feature, the chilled-water system, the fact that cold moderator adds reac-

shutdown time of 1 to 2 hr is regained by cooling the heavy-water moderator before startup. Chilled water from a large insulated tank is recirculated through the heat exchanger in place of water being recirculated through the cooling tower. With the chilled-water loop receiving the heat, the reactor is brought up to full power level and operated in this manner until the temperature in the chilled-water tank is the same as the water from the cooling tower; the heat load is then transferred to the regular loop. During the

time of connection to the chilled-water system, the moderator, by gradually warming up, takes away reactivity from the reactor as fast as it is gained by burn-out of the xenon by neutron absorption. Although the system affords a means for reactivity compensation without having an additional uranium allowance, it is not presumed

EXPERIMENTAL FACILITIES

Access to the neutrons is provided from all eight sides and from the top of the reactor. A typical hole has a thimble at the inner end to isolate the experimental volume from the rest of the pile. A removable shield plug matches the

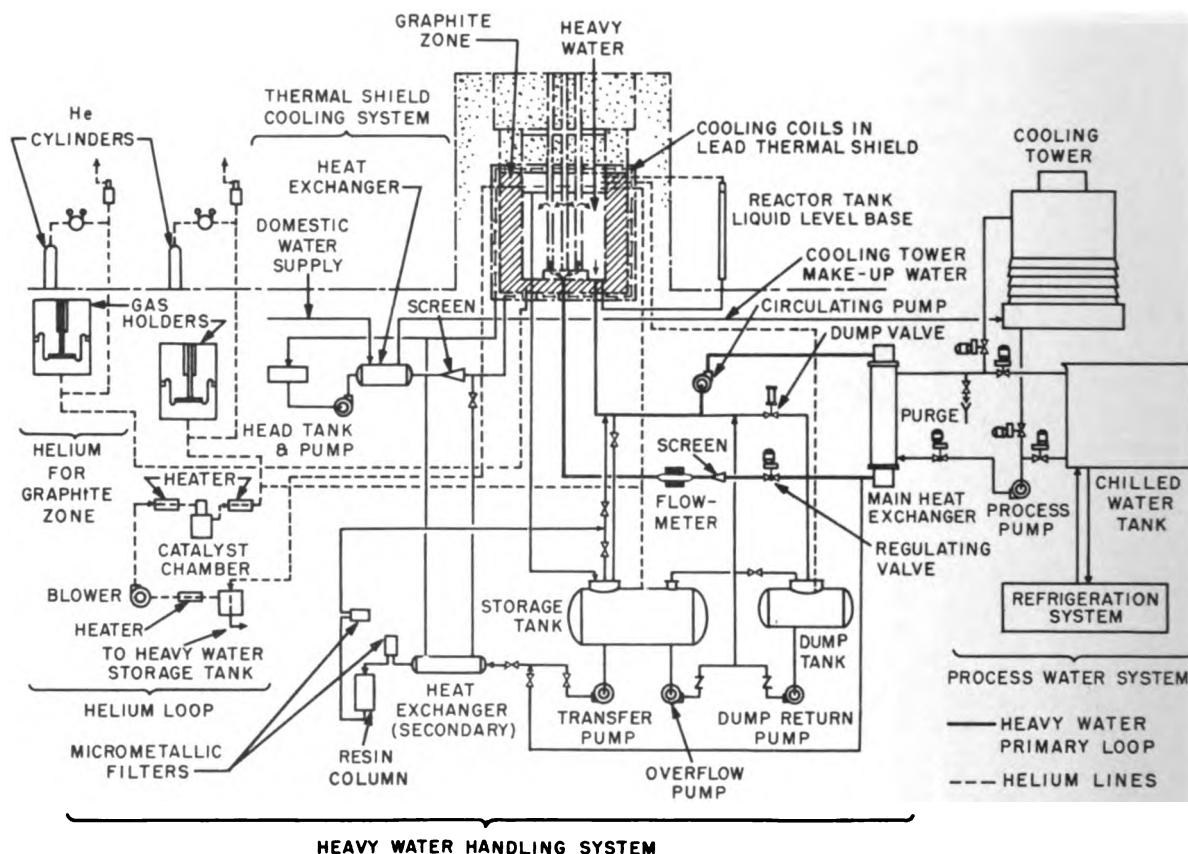


Fig. 5-6 Flow sheet of water systems.

to be a complete adjustment for all the variations in reactivity due to xenon.

The instrument holes complete the features at the reactor, which relate to control. There are eight of these horizontal penetrations of the graphite, four near the top of this zone and four near the bottom. They are equally distributed around the core to give an average instrument signal in the control room, independent of control-rod location. Openings at the reactor faces for the control instruments are near the corners of the octagon, out of the way of most experimental areas.

bulk shield in content and thickness. For a particular experiment the shield plug is replaced with a plug of special design. For irradiation of samples in the thimble volume, the plug might have tubes for cooling fluid and for thermocouples. For studies in which neutrons are allowed to emerge from the reactor, the plug would have carefully machined slots to define a neutron beam.

The main beam holes approach as closely to the core as feasible in order to obtain maximum usefulness of the reactor for beam work. There are four of these facilities; two are made 4 in. in

diameter to reach the region of fast neutrons, and two are made 11 in. in diameter to give the greatest number of beam neutrons. These are designated H-1 through H-4 in Fig. 5-3. Each facility is provided with a rotating plug 18 in. thick for shutting off the beam. Also shown in Fig. 5-3 are four auxiliary beam holes entering the region where the thermal-neutron intensity is still high; these have 12-in.-thick vertical gates for shutting off the beams. All eight of these holes point to the mid-plane of the core.

Another kind of facility, shown in Fig. 5-3, is the neutron column. The neutrons are allowed to diffuse toward the face of the reactor an additional $2\frac{1}{2}$ ft in a 5-ft-square extension of the graphite zone. There are two such extensions, or columns, in CP-5. These are of general utility for large irradiations, for beam work, or as a large-area source of thermal neutrons. Such large sources have been valuable for neutron-diffusion experiments; the results have been utilized in studies of proposed reactor cores and shields. In CP-5, removal of the massive shield for the two graphite extensions is compensated for by special face shielding; a preponderance of lead is used since the reactor gamma rays are scarcely affected by $2\frac{1}{2}$ ft of graphite. In each face shield are two 12-in.-thick gates, or shutters, which move horizontally. Each gate is four segments high, with separate movements. Thus, a pair of gates can be actuated to form openings 4, 8, 12, or 16 in. high in any width up to 16 in.

The neutron columns and the eight beam holes previously mentioned pertain to 6 of the 8 vertical faces of the reactor. The two remaining faces are used for the openings extending completely through the reactor. The lowest facilities are two 8- by 12-in. channels for isotope production irradiations. Above these are two tubes in a plane just below the core. These tubes are connected to pneumatic-transfer systems for transmitting short-life irradiated samples to counting facilities with a minimum of delay. Finally, there are two 6-in. holes, one tangent to the core at a lower edge and one tangent at an upper edge. With these holes it is possible, for example, to study gamma radiations produced by neutron capture in samples of nuclei without interference from extraneous radiations originating in the various reactor parts.

Typical values of thermal-neutron intensities to be expected at a 1000-kw heat-removal rate are given in Table 5-1.

Table 5-1 Typical Thermal-neutron Intensities to Be Expected

	n/cm ² /sec
1. At thimbles just external to fuel zone....	2×10^{13}
2. At thimbles out in heavy water.....	10^{13}
3. At inner thimbles in the graphite.....	2×10^{12}
4. At outer thimbles in the graphite.....	8×10^{11}

The thimbles are arranged in four concentric circles designated 1, 2, 3, and 4 in Fig. 5-3.

REACTOR ARRANGEMENT

The over-all arrangement which permits the experimental approaches just discussed is shown in Fig. 5-1. The reactor is at the center of a circular room 70 ft in diameter, which is reserved entirely for research equipment and auxiliary shielding (Fig. 5-2). The overhead crane system includes a 20-ton main hoist and a 5-ton auxiliary. The room is at the main floor level and has a truck entrance to meet the needs for large and heavy handling operations. The 2-ft-thick concrete floor can support most heavy loads without special consideration; it also acts as a shield between the heavy-water equipment in the basement and the experimental area above.

The reactor operators have a view of the reactor and its surrounding area from the control room on the second floor. A circular catwalk extending around the wall of the reactor room has three staircases to main-floor areas with restricted approach because of neutron beams. The top of the reactor is reached by a bridge that can be positioned at two places along the catwalk. This freedom in shifting the bridge facilitates handling operations with the crane. The cables between the reactor and the control room are in a conduit buried in the main floor.

The reactor itself, the heat-removal system, and the control system are the three salient features which occur in all reactor facilities. Further details about the reactor, including structural arrangement, cooling and thermal-expansion provisions, and shielding considerations are shown in the drawings in the Appendix.

HEAVY-WATER HANDLING SYSTEM

Piping and equipment carrying the heavy water are in an isolated part of the basement (Fig. 5-7) to guard personnel against residual radiation and

carried in stainless-steel piping and equipment except for the aluminum reactor tank and its pipe extensions to the tunnel in the pedestal. Included in the handling system are the primary loop, which removes the heat from the reactor; the secondary loop, used for heavy-water puri-

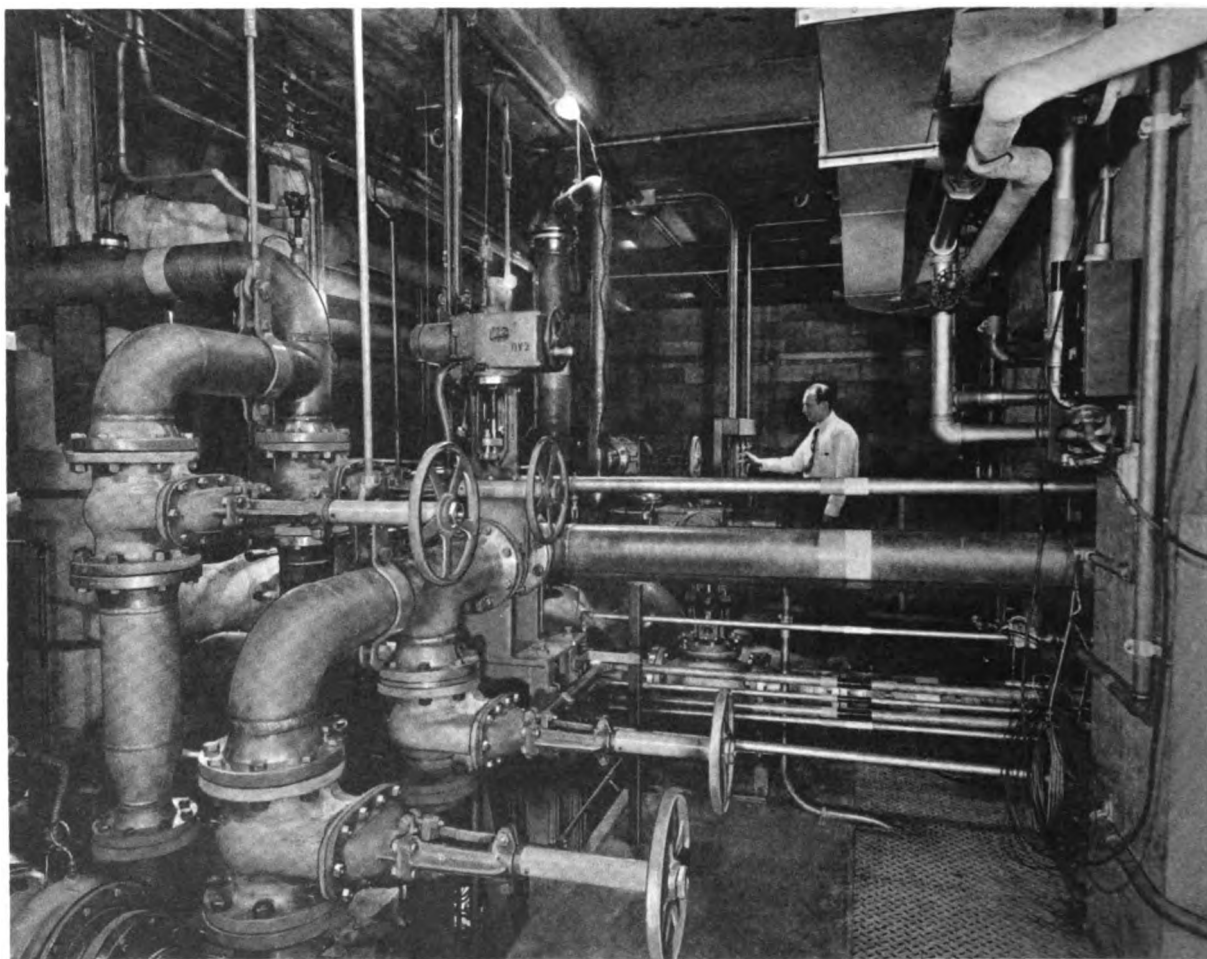


Fig. 5-7 Part of D_2O system.

to guard the equipment against unauthorized manipulation. Two feet of concrete shielding is provided for protection against radiation from possible impurities in the water and from nitrogen 16. The deuterium of the heavy water produces a minute amount of tritium under irradiation, which is of possible concern if some of the heavy water were spilled. Ventilation is provided in the isolated area as a precaution.

The heavy water, amounting to 15,000 lb, is

fication; the storage tanks; and a gas-phase piping system. This last system, employing helium, is used for balancing pressures and for the recombination of a small amount of deuterium and oxygen. In the primary loop the 6-in. mains are arranged as compactly as is feasible in order to reduce heavy-water inventory. The shell-and-tube heat exchanger contains $\frac{3}{8}$ -in.-diameter tubes on $\frac{1}{2}$ -in. triangular pitch for the same reason. The pressure in the 731 tubes in the heat

exchanger is maintained higher than on the shell side so that any leak which might develop would result in losing only a small fraction of the heavy water rather than almost complete loss through isotopic dilution.

The water-systems flow sheet (Fig. 5-6) indicates the major equipment pieces and the pipelines for normal flow. The only points of proximity of light and heavy water are in the main heat exchanger and the secondary heat exchanger. The latter cools the heavy water prior to contact with the resin columns, which remove ionic impurities. The gas holder in the helium loop is of the diaphragm type to give assurance that hydrogen will not enter the system through an oil seal.

There are several miscellaneous parts to the heavy-water handling system in addition to the major items indicated on the flow sheet. All these major pieces, including the large piping, are connected to a small tank by a number of pipes to drain the heavy water before opening up any part of the system. There also is a leak-detection system, which gives an alarm if a drop of water falls on a printed conductor grid suspended under a seal point. Since each point has a grid, the problem of finding a possible leak is simplified by having several grid circuits, each with an alarm light. The heavy-water system also has several pressure- and temperature-sensing elements. Some temperature signals are combined with flow for power indication and recording, and some are used to protect the system. For example, there are low-temperature alarms on both the heavy-water and ordinary-water loops since heavy water freezes at 7°F above the freezing point of ordinary water.

COMMON-WATER SYSTEMS

Ordinary water, designated process water on the flow sheet (Fig. 5-6), receives the reactor heat in the main heat exchanger and delivers it to the atmosphere in a forced-draft cooling tower. Make-up water for evaporation and purge extracts heat from the thermal-shield cooling system before going to the tower. The shield-system heat load is only 2 to 3 per cent of the total, including all of the graphite heating. Continuation of shield cooling after shutdown assists in fission-product heat removal; however, shutdown cooling is not essential below 4000-kw reactor operation

since the heat capacities of the heavy water and graphite are enough to store the heat produced over several days.

All valving operations necessary for starting the reactor are performed in the control room. Regulating valves on the heavy-water and the process-water loops and switching valves between the cooling tower and the chilled-water tank are motorized for remote-station actuation. Only special operations, such as switching pumps or changing the direction of the cooling-tower fan rotation for deicing, are locally controlled.

REACTOR-CONTROL INSTRUMENTATION

For the purpose of operating a nuclear reactor, certain information is needed about the status of the reactivity defined and discussed under Control Rods and Related Reactor Features. The sensing elements which follow the chain reaction behavior are ion chambers producing electric signals in proportion to the number of neutrons entering the chambers. The easiest way to make the electric signal into a visual performance is with the aid of a galvanometer. The CP-5 has both a power galvanometer and a differential galvanometer. The latter instrument indicates the difference between the ion-chamber signal and a chosen reference signal in order to follow small changes near a power level corresponding to the reference. Figure 5-8 shows the control-panel layout.

The range of neutron intensity of interest requires adjusting the galvanometer sensitivity over several powers of 10. In fact, when a completely new fuel loading is placed in the reactor, the initial neutron intensity must be created by a source of neutrons, for example, a mixture of radium and beryllium. The vibrating-reed electrometer, the control-room instrument for the CP-5, responds to the very feeble electric signal of a sensitive ion chamber inserted in the reactor to pick up the low intensity of the source neutrons. The range of the electrometer overlaps the range of the galvanometer to permit a continuous signal up to full power.

Information received in the control room is acted upon by closing circuits to the drive motors of the control rods or deenergizing the magnetic clutches if shutdown is desired. Shim-safety rod

movement is followed by a dial giving the angle of rotation; each rod has a dial to give continuous position indication. The regulating-rod position is given directly to a tenth of a centimeter by a register.

The galvanometer setup, the motor control buttons, the position indicators for the control rods, and a stop watch meet the needs of the operator in normal startup of the reactor. Two shim-

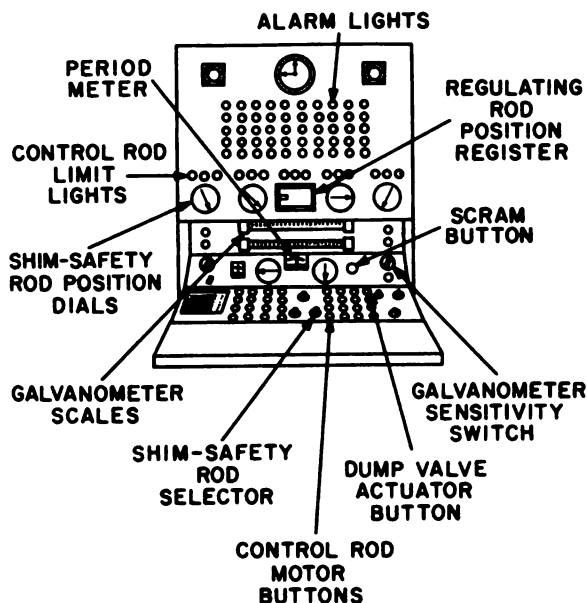


Fig. 5-8 Layout of control panel.

safety rods are withdrawn to the predetermined out limit to be poised for safety movement. The other two are moved out slowly to intermediate positions below the locations at which the critical condition of the reactivity equal to unity is expected. During these movements, the neutron intensity increases with rod movement but levels off when the rod stops. The critical condition is approached by causing the regulating rod to explore a small change in reactivity ahead of the shim rods and then moving a shim rod out to correspond to the reactivity of the regulating-rod travel; the latter rod is returned to a position near its in limit before the shim-rod movement. Criticality is reached when it is noted that the galvanometer light continues to move after the regulating-rod movement is stopped. Since the regulating rod cannot cause neutrons to multiply any faster than is determined by the time charac-

teristics of the delayed neutrons, the rate of the increase will at most be a factor of 3 every 20 to 30 sec. The stop watch is used to give a sufficiently accurate value. The regulating rod, or a shim rod, can be moved in slightly to bring the reactivity down to unity and thus place the reactor on steady neutron intensity.

The operator is assisted in several ways by other instruments. A period meter gives a direct reading of the time required for the neutron intensity to change by a factor of $e = 2.72$. For steady-state operation the period is without any limit, as is shown by the infinity sign on the period-meter scale. Another instrument indicates the reactor power level obtained from temperature change and flow rate of the heavy water; these measurements give the most reliable indication of reactor power. Steady power can be maintained automatically by the regulating rod moving up and down, keeping the neutron-intensity signal within 0.5 per cent of a chosen reference in the range from 10 per cent to full power. If the regulating rod reaches a limit, the operator moves a shim rod and the automatic control shifts the former rod toward its mid-point.

The regulating rod on a limit position is an example of several mild alarm conditions which are brought to the operator's attention by indicator lights. The system also has several interlocks between information coming to the control room and the shutdown, or scram, circuit so that the reactor is automatically shut down if certain limits are trespassed. As an example, there are three power-level trips; these are tested periodically. As another illustration of mandatory shutdown, certain experiments in the reactor require one or more scram interlocks to protect the experiment or the reactor.

The control system also includes protection against ambiguity or cross-purpose operation. For example, a selector switch connects only one shim-safety-rod motor at a time to the power source. All of the protective features are to ensure operation within the chosen limits.

REACTOR STRUCTURE DETAILS

The reactor, weighing 600 tons, rests on a concrete pedestal with its spread footing, in turn, supported on undisturbed clay. The reactor

shield extends downward into the pedestal in order to complete the shielding required for the basement areas. The inner parts of the reactor are supported on the accurately machined edges of 3- by 12-in. steel billets embedded in this shield concrete (Figs. 5-9 and 5-10). Loose

A vertical section cut through the middle of the pile is shown in the Appendix (Figs. 5-23 and 5-24).

The top shield plug, also 30 in. thick, has a separate center piece with holes that are accurately aligned with the fuel-assembly holes in the

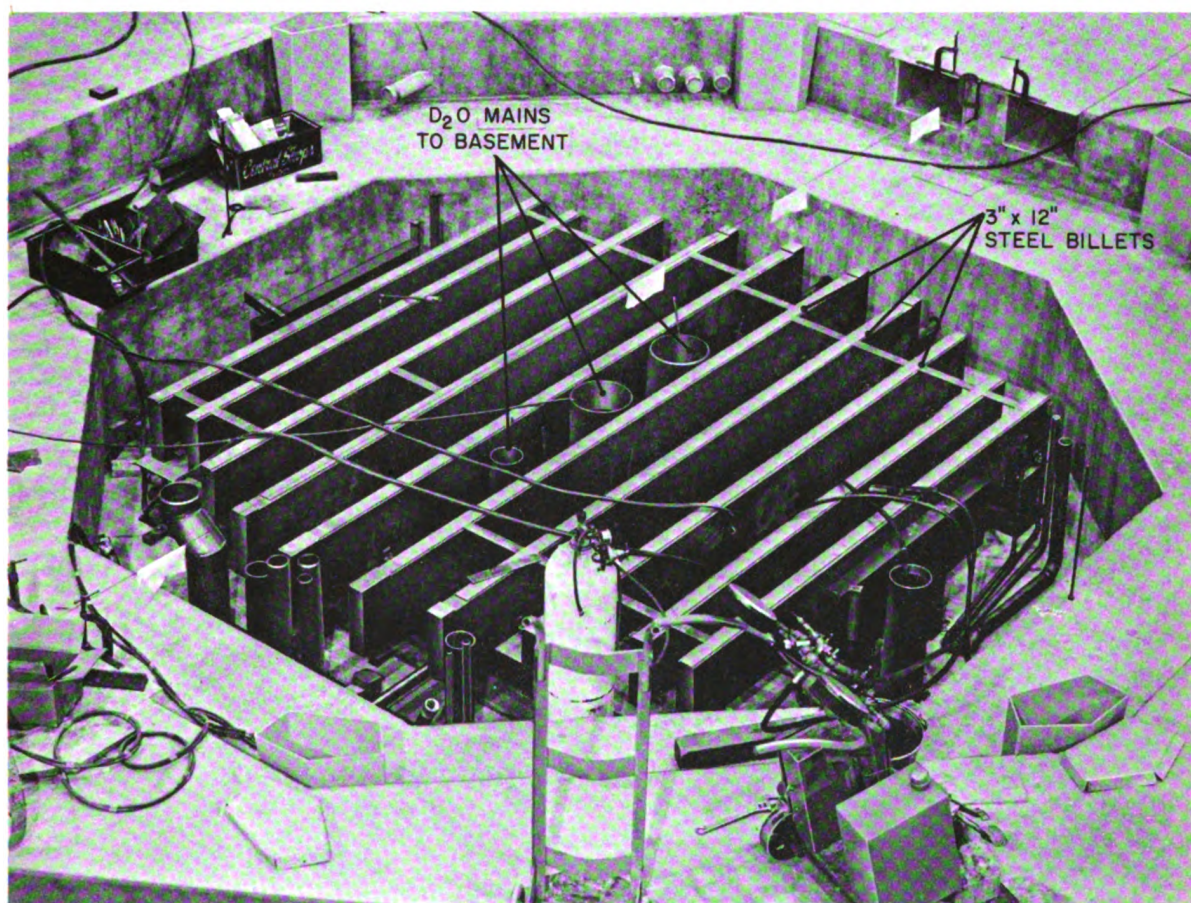


Fig. 5-9 Steel billets.

pieces of boral, boron carbide in $\frac{1}{4}$ -in.-thick aluminum, form a complete cover over the billets. The aluminum tank (Fig. 5-12), with all its accurate relationships to shield openings, rests on machined graphite (Fig. 5-13) laid on the boral. The top section of the aluminum tank and an inner shield plug engage over a height of 30 in. The heavy-water vapor seal is made at the supporting flange at the top of the tank and its plug. An annular shield of 20 tons rests on the graphite around the aluminum tank and completes the first 30 in. of shielding over the experimental volume.

inner shield plug below. The top shield plug rests on the top of a 10-ft-diameter steel shell (Fig. 5-11). Springs in this plug remove about half of the load of the inner shield plug from the aluminum tank. The steel structure of the top shield plug is built up with radial webbing full height and with a 1-in. cover plate plug welded to the webs in order to absorb a possible energy load of the 7-ton fuel-handling coffin.

The graphite and the annular shield just above it are in a gastight section of the steel shell, the top closure being an annular diaphragm of $\frac{1}{8}$ -in.

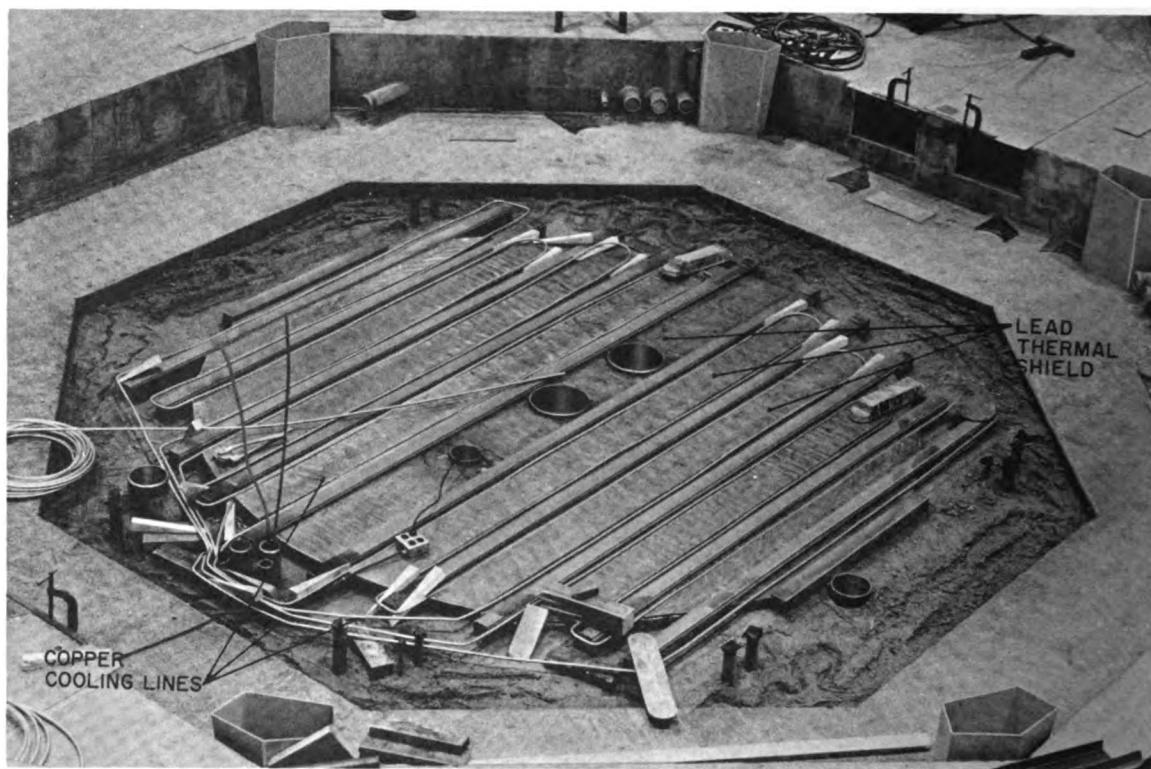


Fig. 5-10 Steel billets with concrete poured.

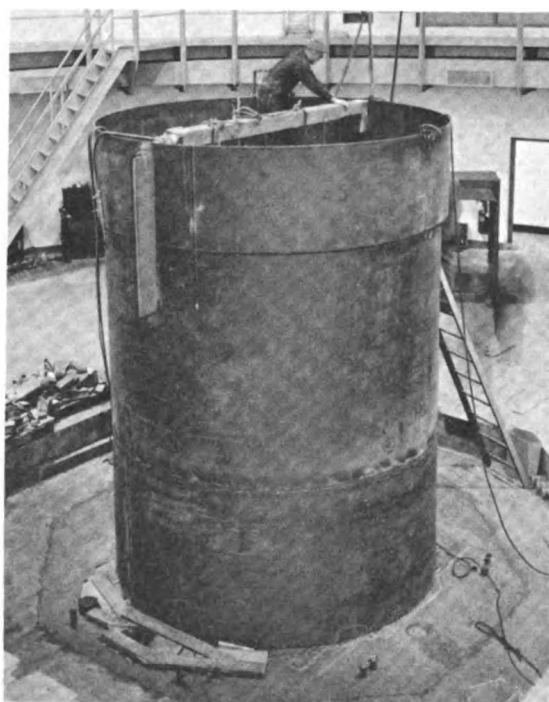


Fig. 5-11 Ten-foot steel tank in place.

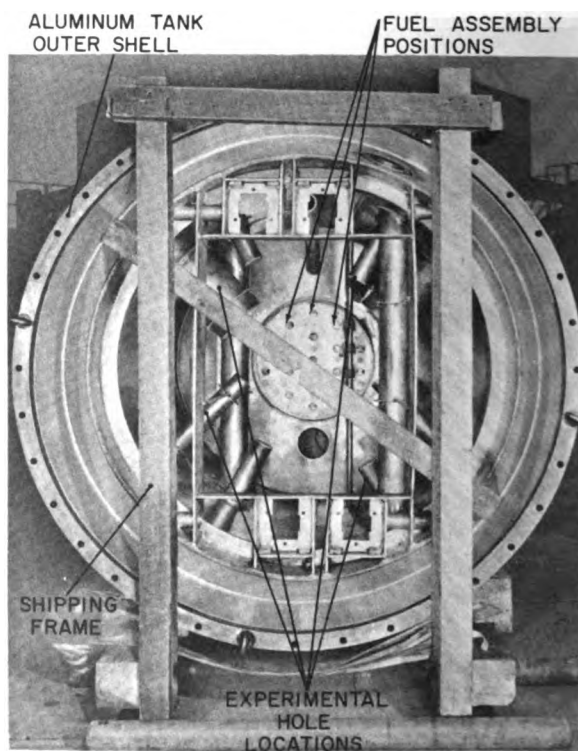


Fig. 5-12 Six-foot-diameter-aluminum-tank interior

stainless steel flanged to the inner shield plug and to the steel-shell offset. The bottom of the steel shell is welded to the billets. Strips of steel welded to recessed surfaces of the billets complete the bottom seal. The openings which penetrate the strips of steel, the steel shell, and the diaphragm are sealed by a variety of methods. The

The graphite is cooled by conduction. Most of the heat ultimately reaches the water-cooled coils in the $3\frac{1}{2}$ in. of lead surrounding the graphite zone. Thus, there are coils in the lead at the bottom of the annular shield taking heat from the top of the graphite and coils in the lead between the billets taking heat from the bottom of the

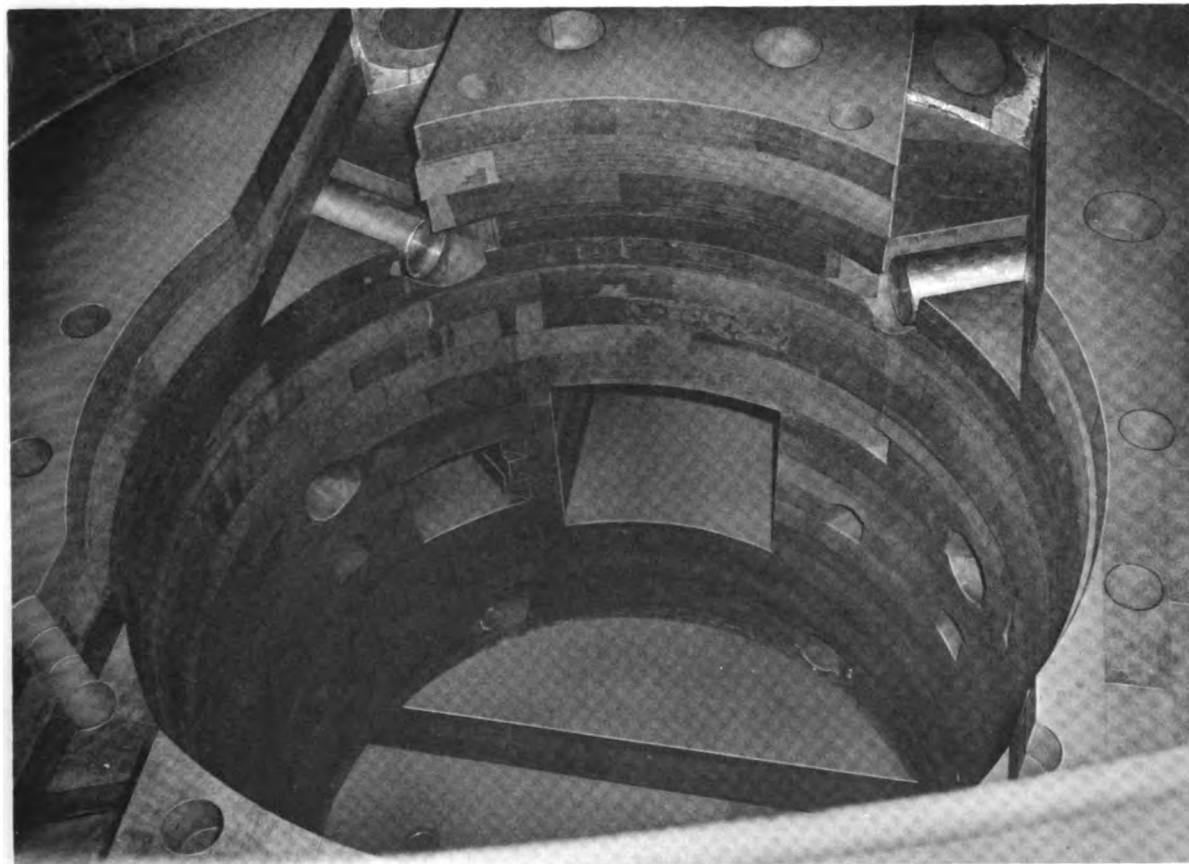


Fig. 5-13 Graphite blocks prior to insertion of reactor tank.

flange on the thimble of an experimental hole is the most generally used detail. Figures 5-9 to 5-22 show the CP-5 in various stages of construction.

Structure Cooling. The graphite zone is in an atmosphere of helium so that the parts of experimental equipment near the reactor core can be cooled, in many instances, by heat conduction across a helium gap between the thimbles and the surrounding tube sections, which form parts of the aluminum tank holding the heavy water.

graphite. The heat from the central portion of the graphite reaches the coils outside the steel shell by traversing lead fill, $\frac{1}{4}$ -in. boral, $\frac{1}{2}$ -in. steel, and finally lead poured around the coils.

There are cooling coils in the lead at the bottom of the inner shield plug about $3\frac{1}{2}$ ft above the reactor core. The weighting plugs of the fuel assemblies are cooled by a close fit to the inner shield-plug liners. Shim-safety-rod shafts and bearings are cooled by conduction through the signal arms, which always dip into the heavy water.

Thermal Expansion. Thermal expansion of the aluminum tank flexes the $\frac{1}{8}$ -in. stainless-steel diaphragm over the annular shield; bellows between the annular shield liners and the diaphragm provide seals and expansion joints at vertical holes penetrating the graphite. The annular shield itself rides up and down as the temperature

allowed for at the inner engagement; outward expansion against a spring is also provided for in the design. The fuel assemblies have a several thousandths of an inch more upward movement than the aluminum tank and its plug. This differential is taken up by the O-ring seals on the weighting plugs.

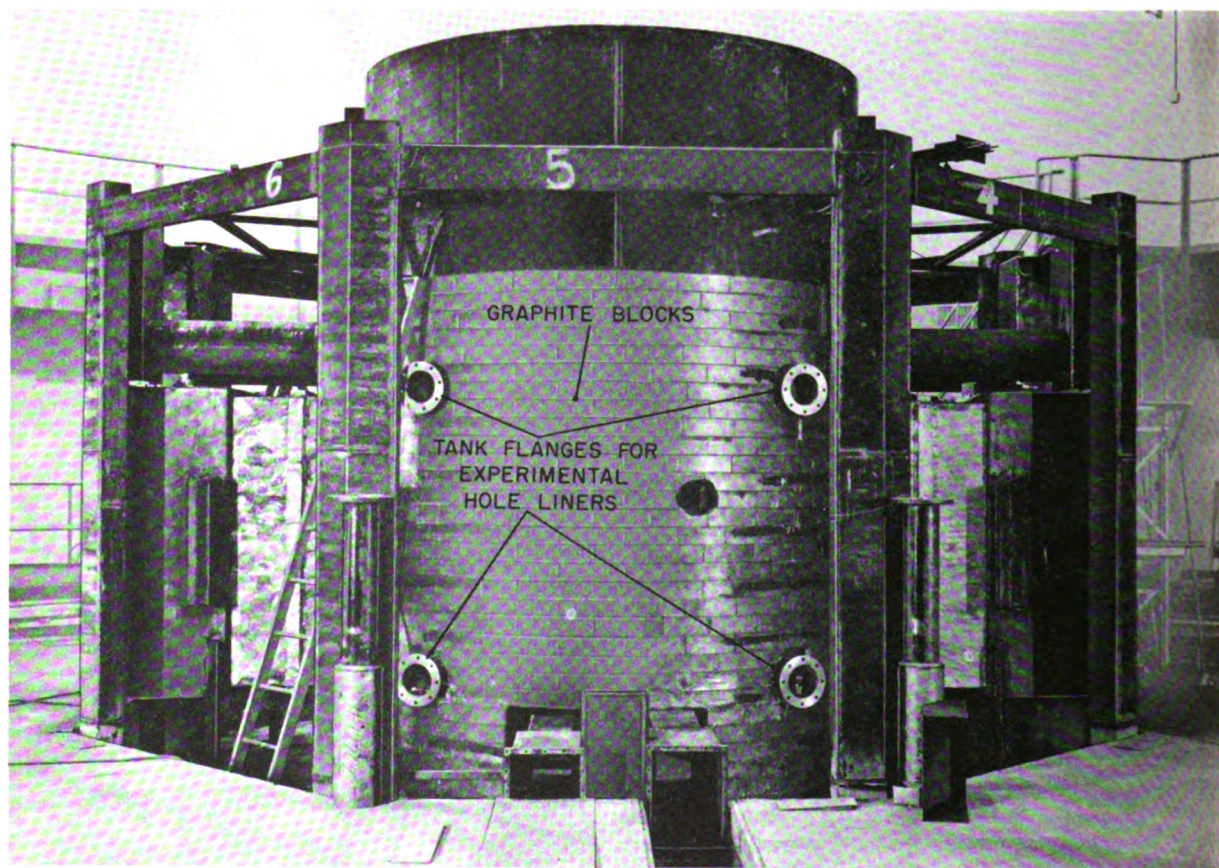


Fig. 5-14 Steel-tank flanges placed for experimental-hole liners.

of the graphite changes. Thimbles of horizontal experimental holes expand inward; their walls are $\frac{1}{16}$ in. thick to allow the thimbles to move with the $\frac{1}{16}$ in., or less, vertical expansion of the aluminum tank. Bearings of the shim-safety rods are mounted on a grid structure in the aluminum tank so that allowance is made for the 9-ft shafts to move up and down by means of clearances in the spline type of engagement at the inner end and by means of self-aligning ball bearings near the outer ends. Heating of the shafts by nuclear radiation causes significant expansion which is

Shielding and Further Reactor Details. Outside the regions of significant heat release is the so-called biological shield, constructed of iron-bearing concrete of sufficient thickness to permit people to work around the reactor. Actually, the outer-shield requirement of the CP-5 is based mostly on instrument tolerance. The escaping radiation is far below what is permissible for personnel; it is of the order of instrument detection to minimize pickup by experimental counters. Some orientation on the performance of the total shield may be pieced together by noting that

7×10^{16} neutrons are created every second at 1000 kw. Roughly one-half escape the fuel zone and only the order of 10^6 per second escape the reactor. Thus, the number absorbed is so close to the number leaving the core that the most convenient way to express the shielding ability seems to be to state that the neutron attenuation

spaces within the outline of the octagonal prism afford locations for recessing auxiliary ducts and equipment. Thus, a shielding setback at the top of the pile provides an enclosed shelf 2 ft deep and 3 ft high for a dozen drive packages which actuate gauges and shutters on experimental openings. Junction boxes, thermal-shield piping mani-

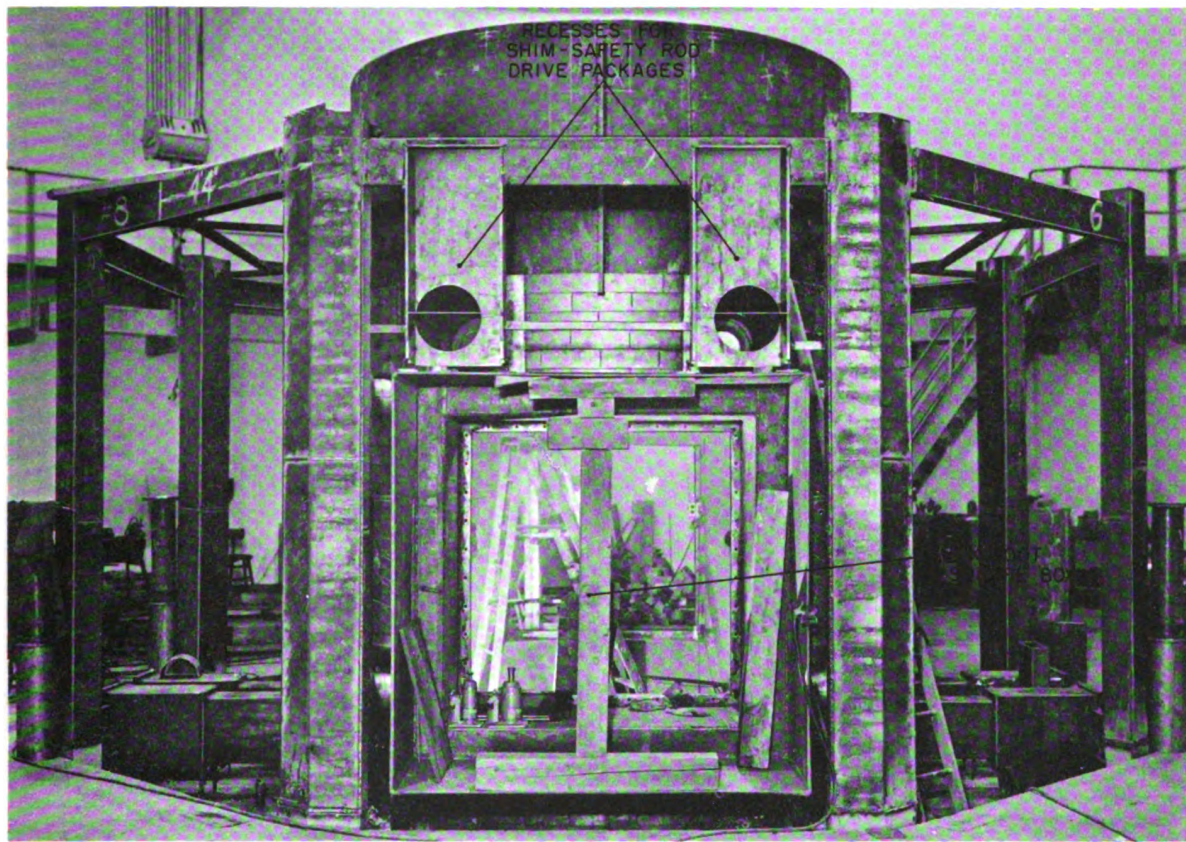


Fig. 5-15 View along axes of neutron column.

is of the order of 10^{16} . There is a somewhat smaller factor for the gamma-ray attenuation.

The outer shield of 56 in. is slightly thicker than would be required for a solid chunk of material. The many penetrations of the shield with their experimental provisions amount to imperfections for scattered radiation to bypass a small thickness of the bulk shield.

The outside of the reactor is not the exact shape of required shielding. A better picture is obtained by imagining a cylinder with slightly rounded ends, similar in shape to a pressure tank with the diameter equal to the height. The extra

folds, and vertical experimental-hole cooling lines are also concealed in this space. The continuations of these cooling lines to the vertical experimental holes are laid in ducts under the top of the pile. The reactor is about 6 in. higher than a height based on shielding the fuel zone, allowing for the ducts and a disk of lead shielding for safe withdrawal of a spent fuel assembly. This added reactor height also makes the enclosed shelf more spacious.

The extra spaces at the corners of the octagon are utilized as vertical air ducts drawing 6000 cfm of room air toward the reactor; the number of air

changes increases as the reactor is approached in correspondence with the increased chance of detectable air activity. The thickness allowance in the biological shield for imperfections is utilized by providing pipes for an internal venting system

for each of the main experimental openings. With normal precautions this system prevents escape of activity into the room, especially when air is displaced from a thimble during exchange of experimental plugs.

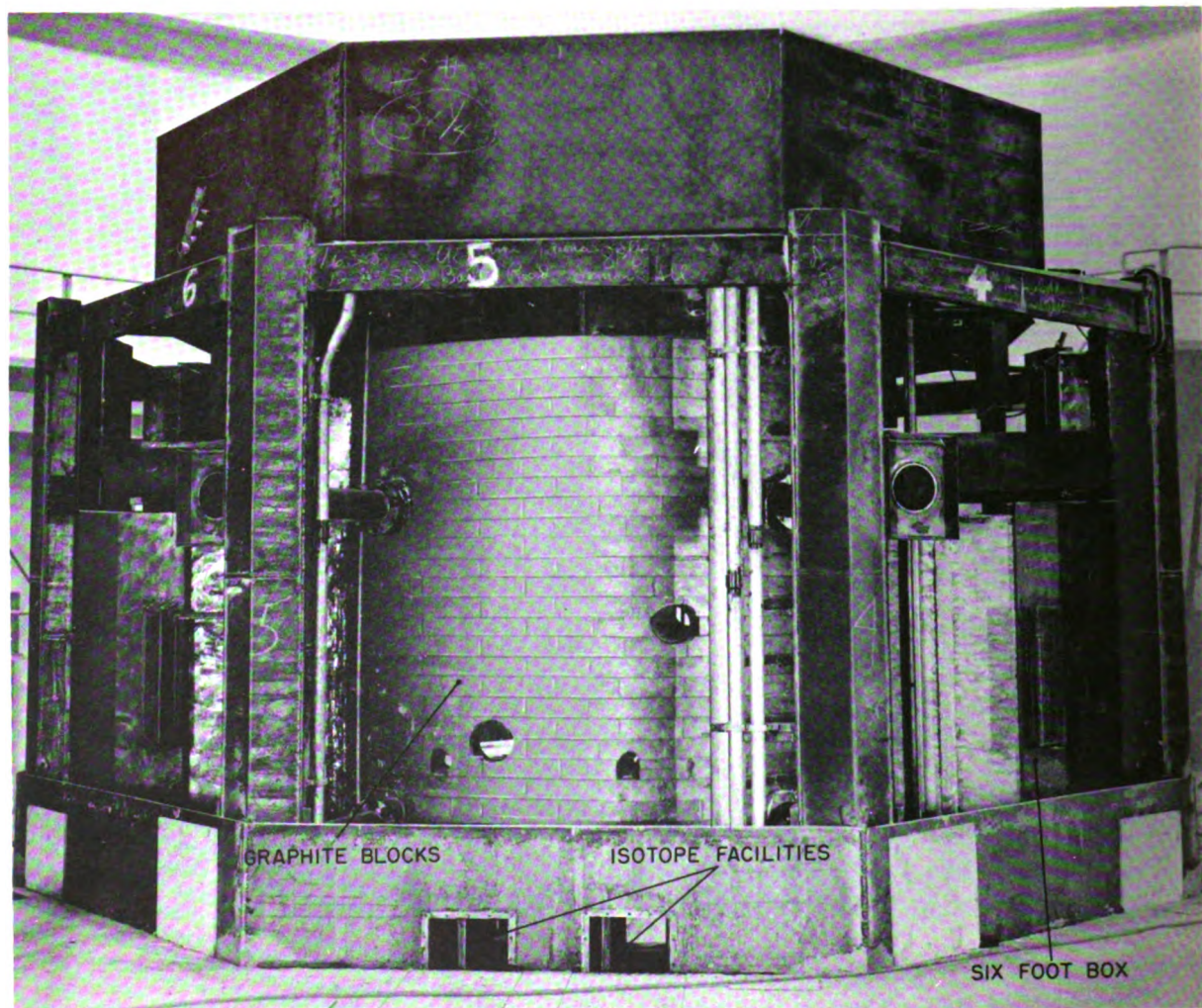


Fig. 5-16 View with concrete poured to floor level.

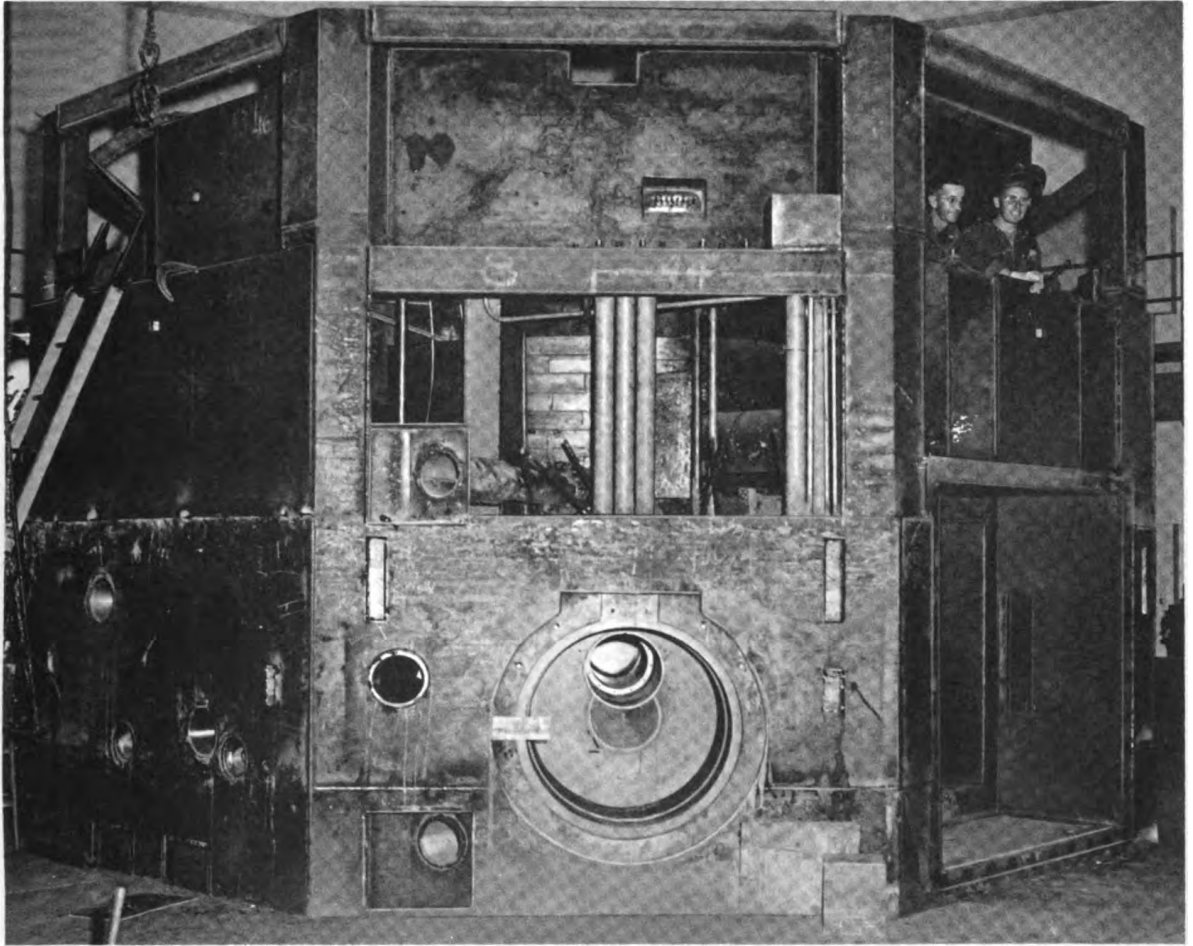


Fig. 5-17 Experimental hole without rotating plug.

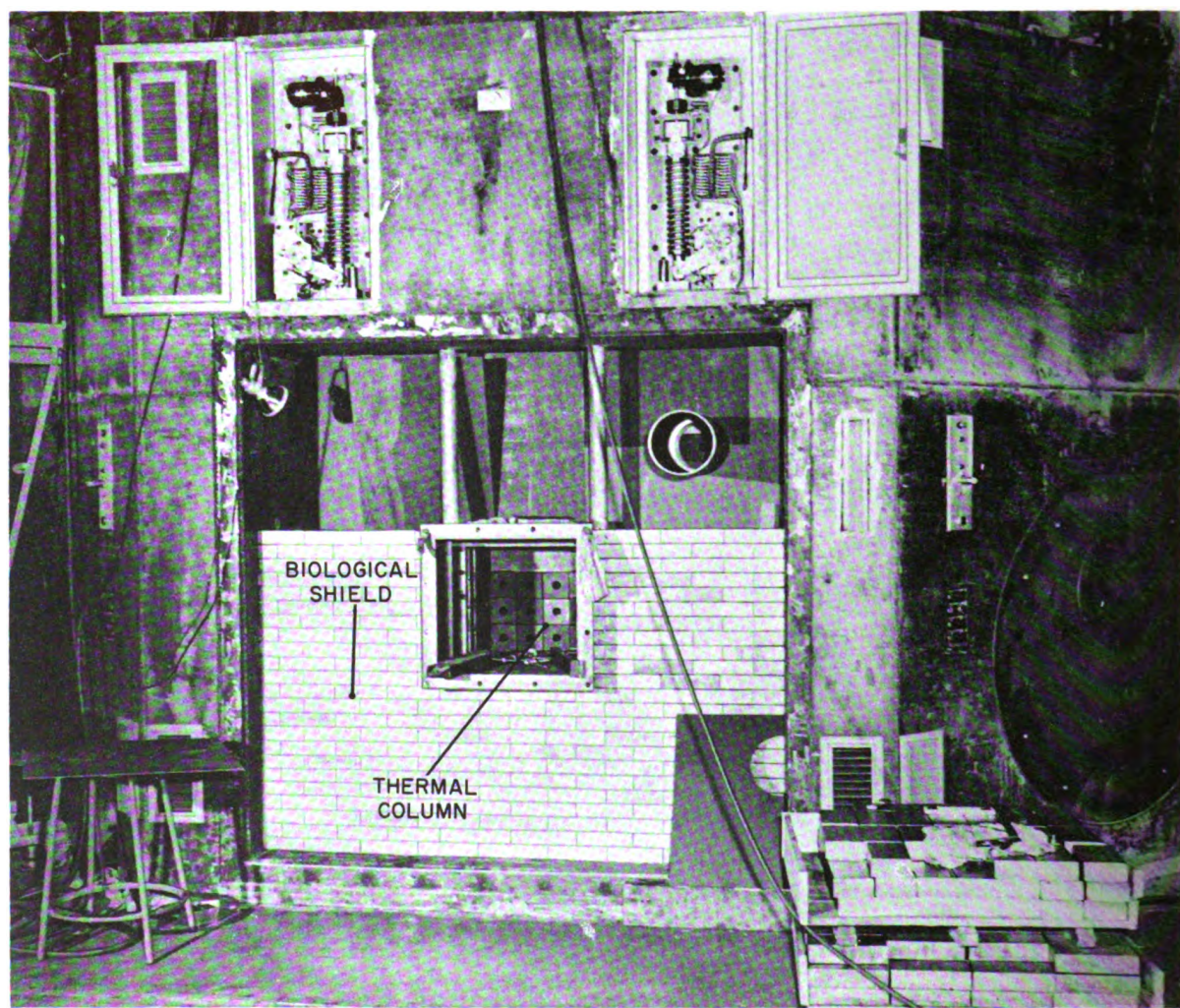


Fig. 5-18 Thermal column, front view.

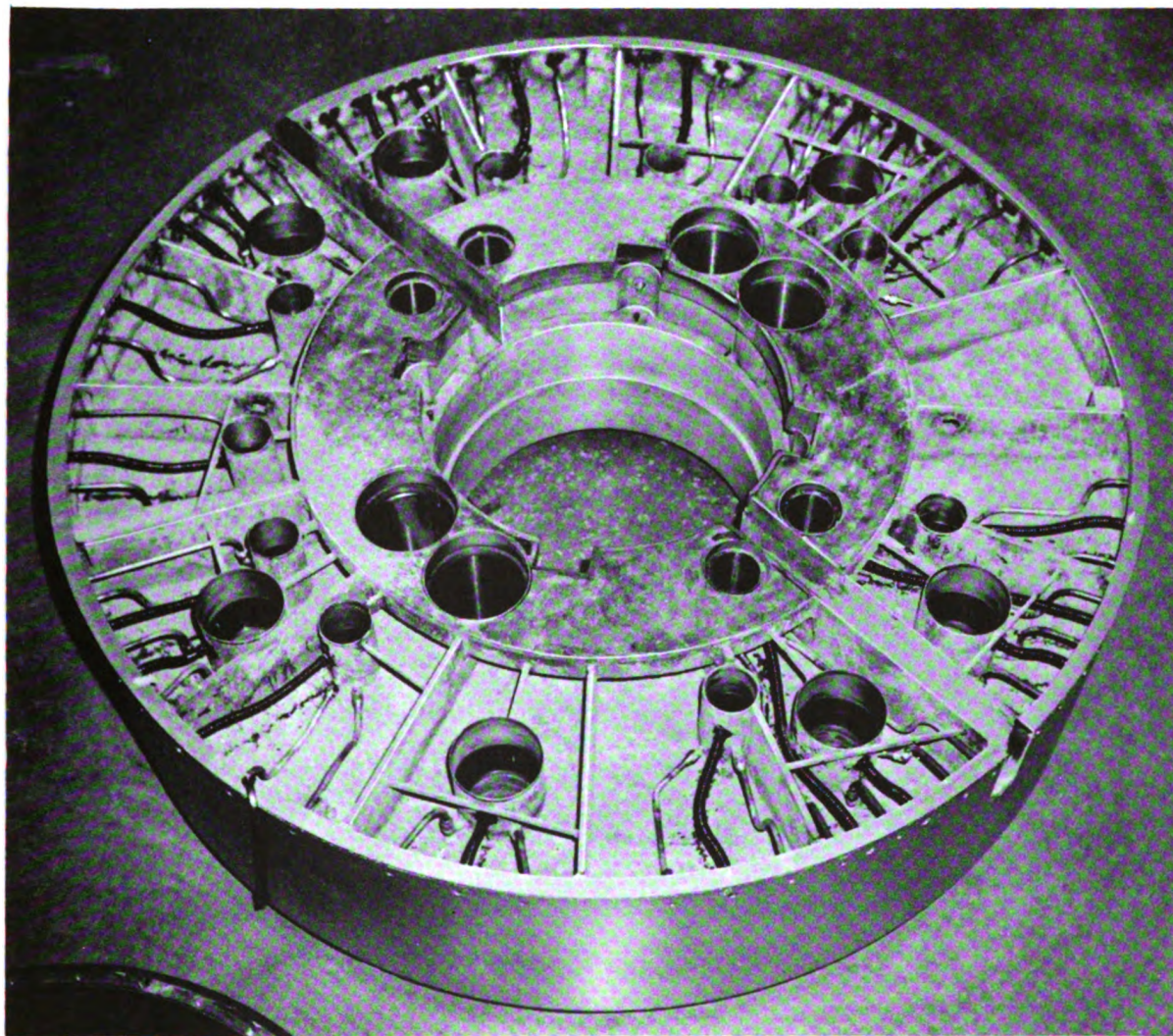


Fig. 5-19 Outer top shielding plug, before insertion.

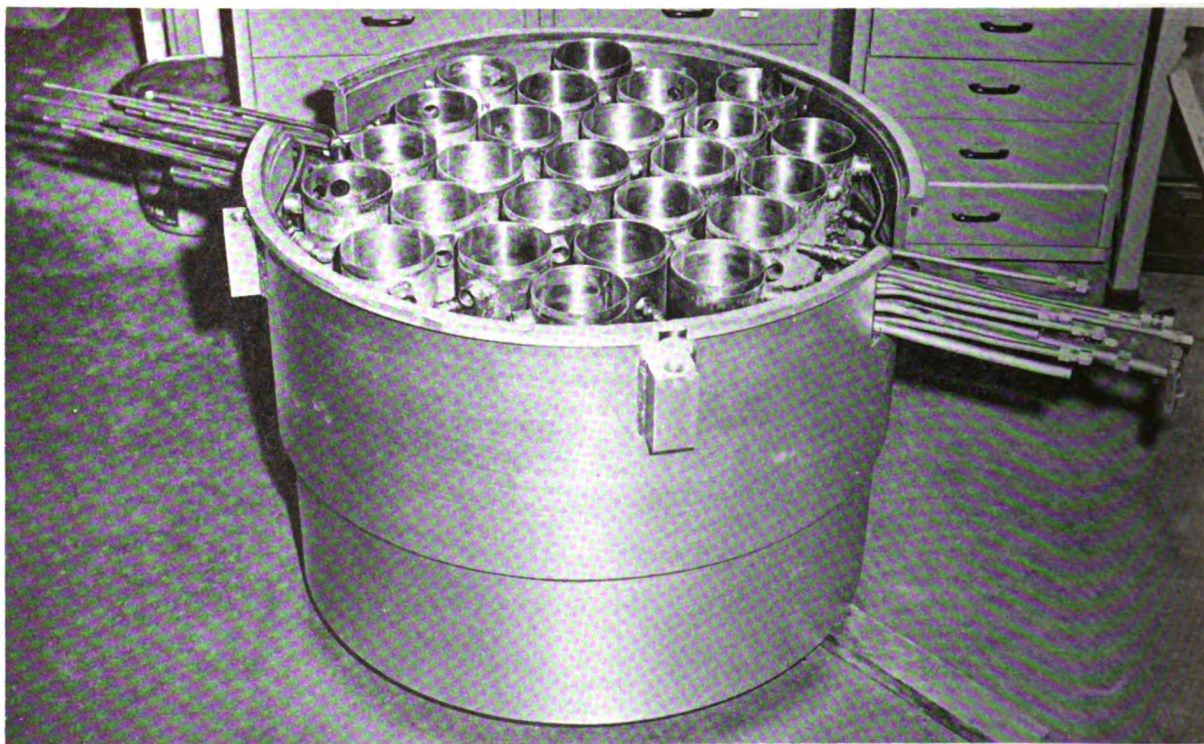


Fig. 5-20 Inner top shielding plug, before insertion.

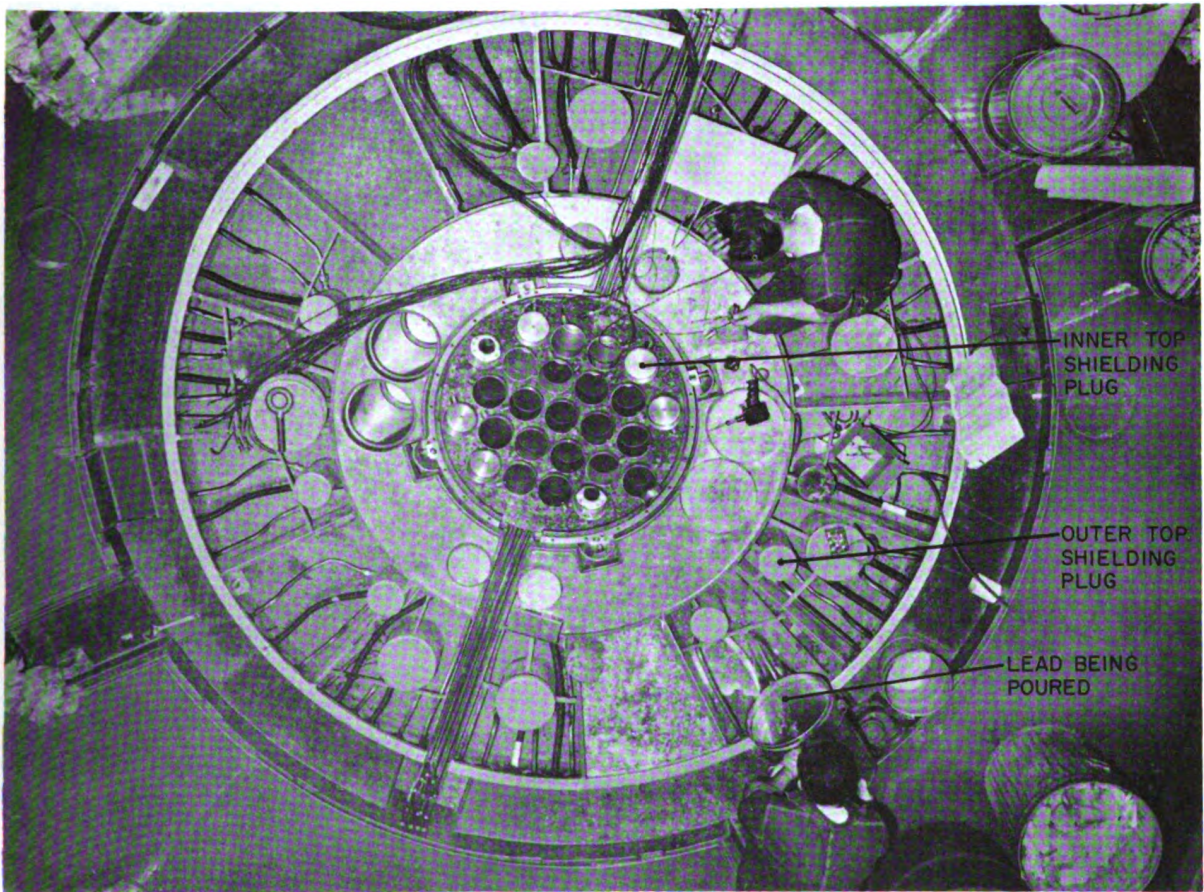


Fig. 5-21 Reactor, before addition of cover plate and lead shielding.

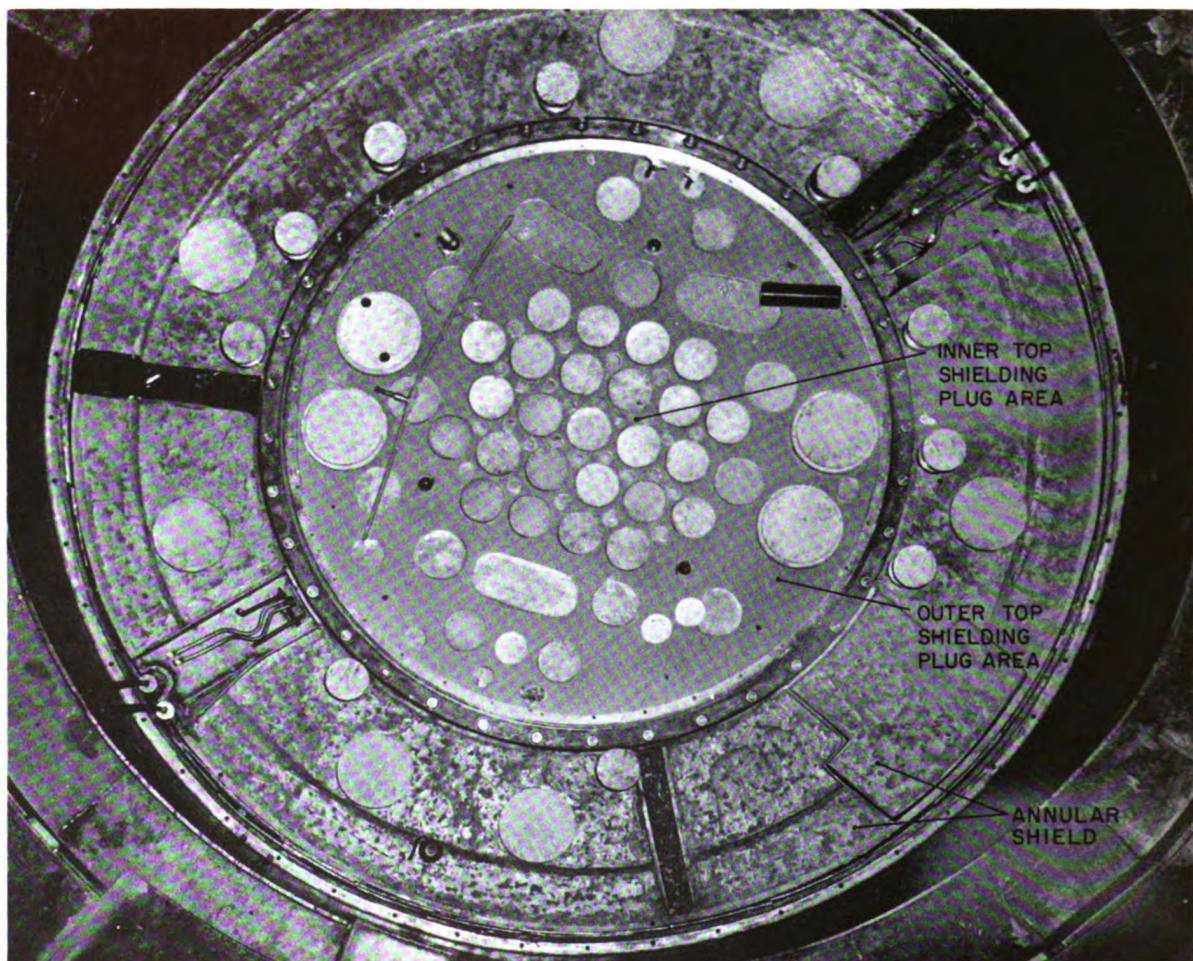


Fig. 5-22 Reactor, after addition of cover plate and lead shielding.

APPENDIX

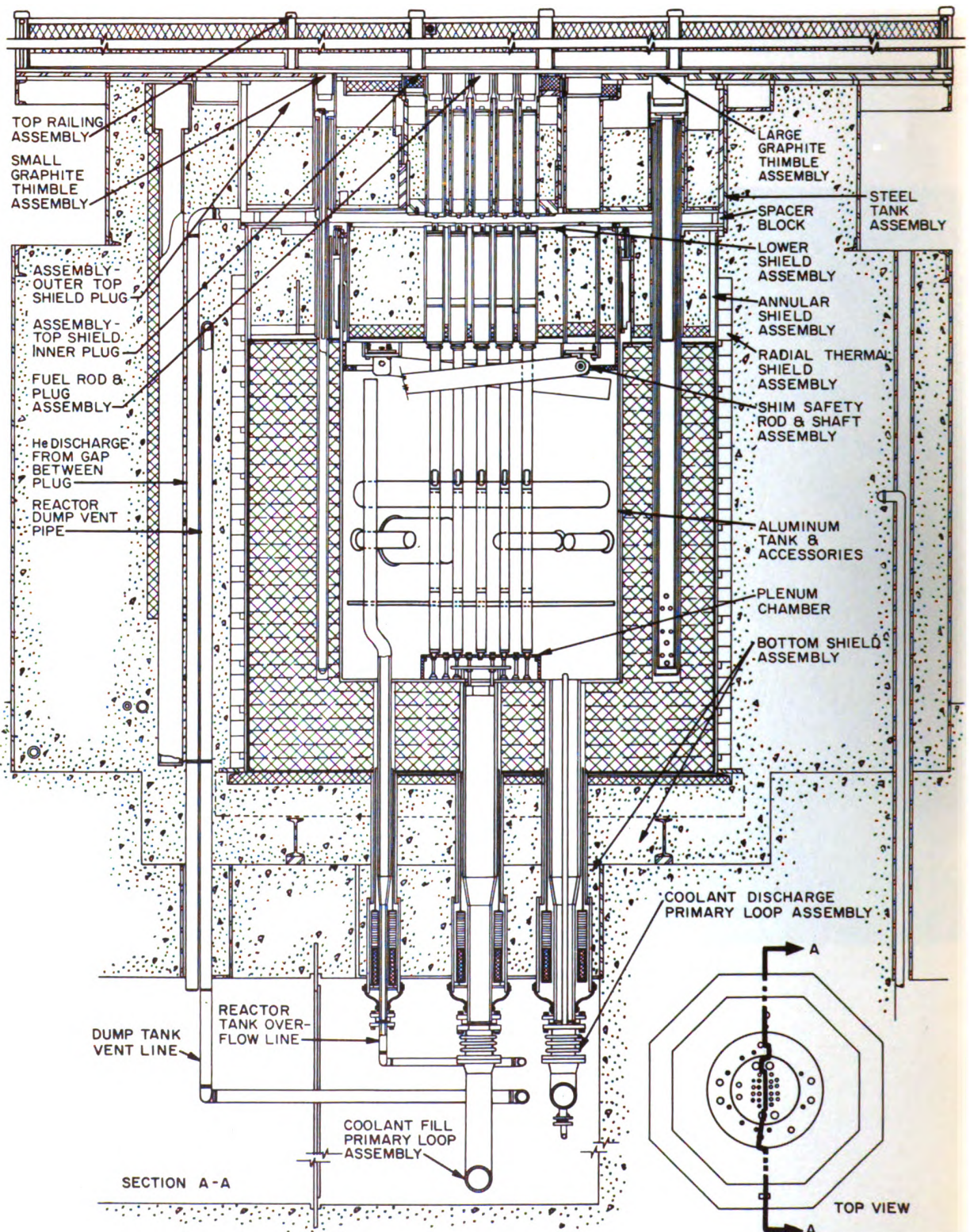


Fig. 5-23 Vertical section.
356

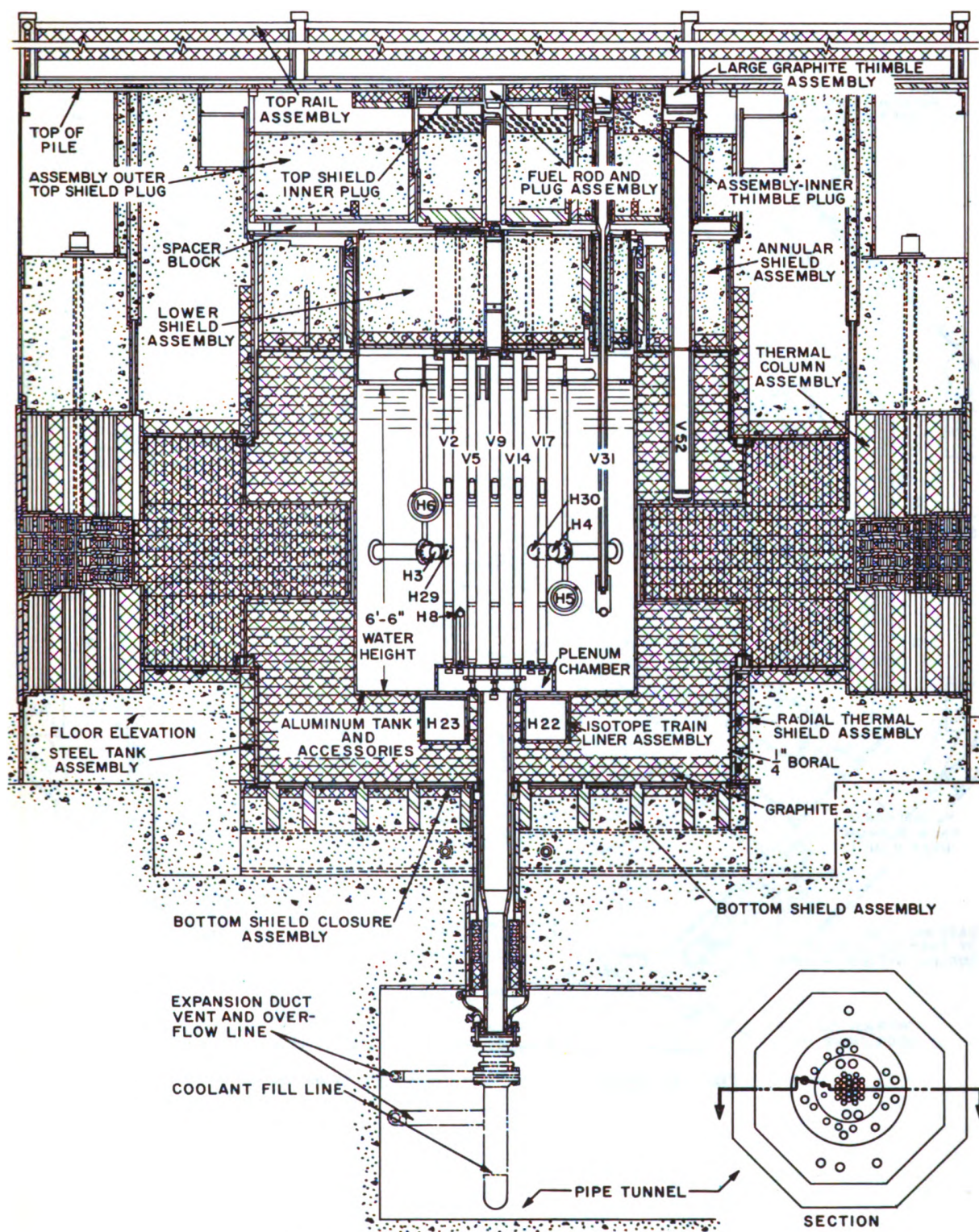


Fig. 5-24 Vertical section through thermal column.

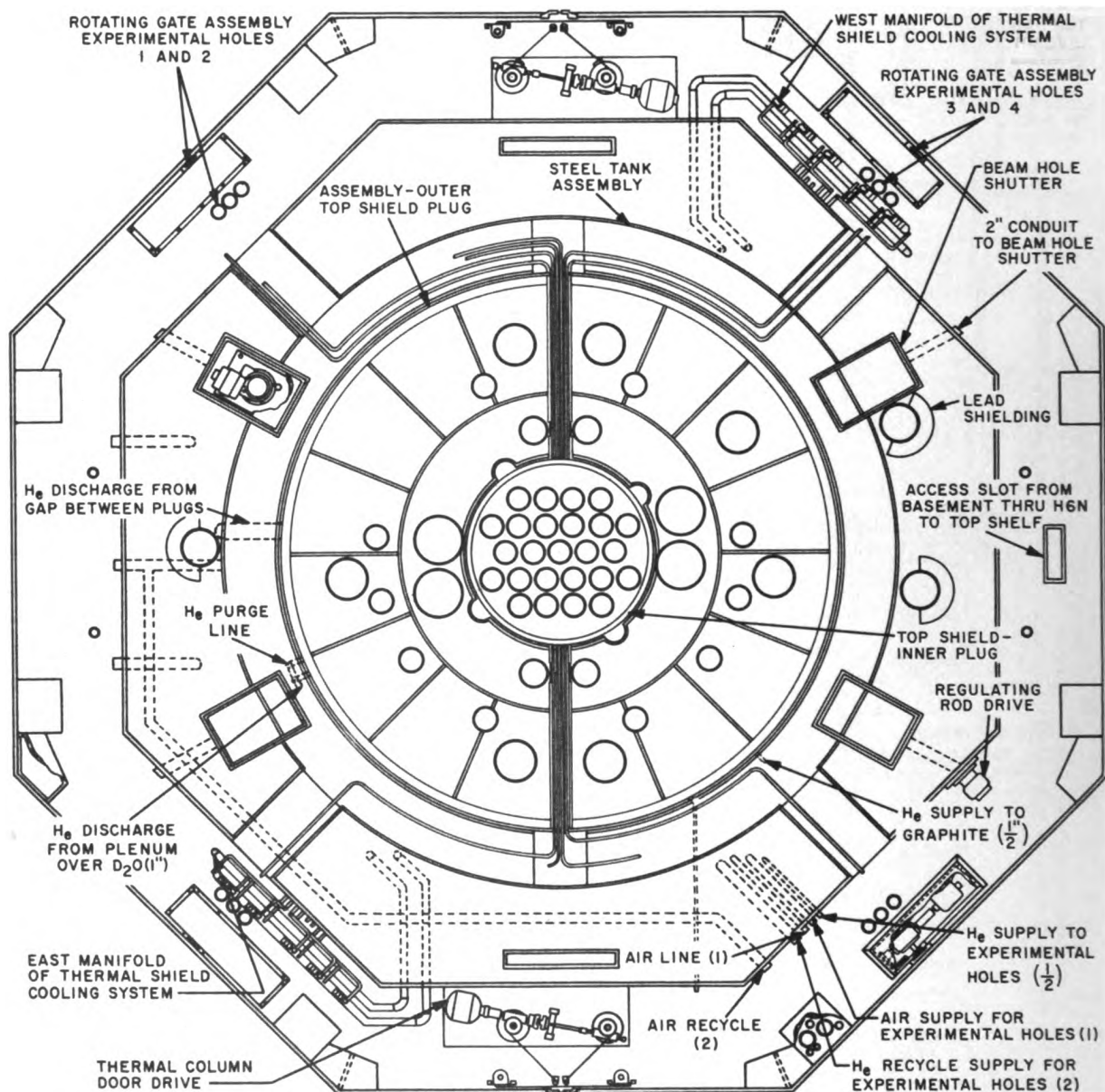


Fig. 5-25 Horizontal section with 1-in. top plate removed.

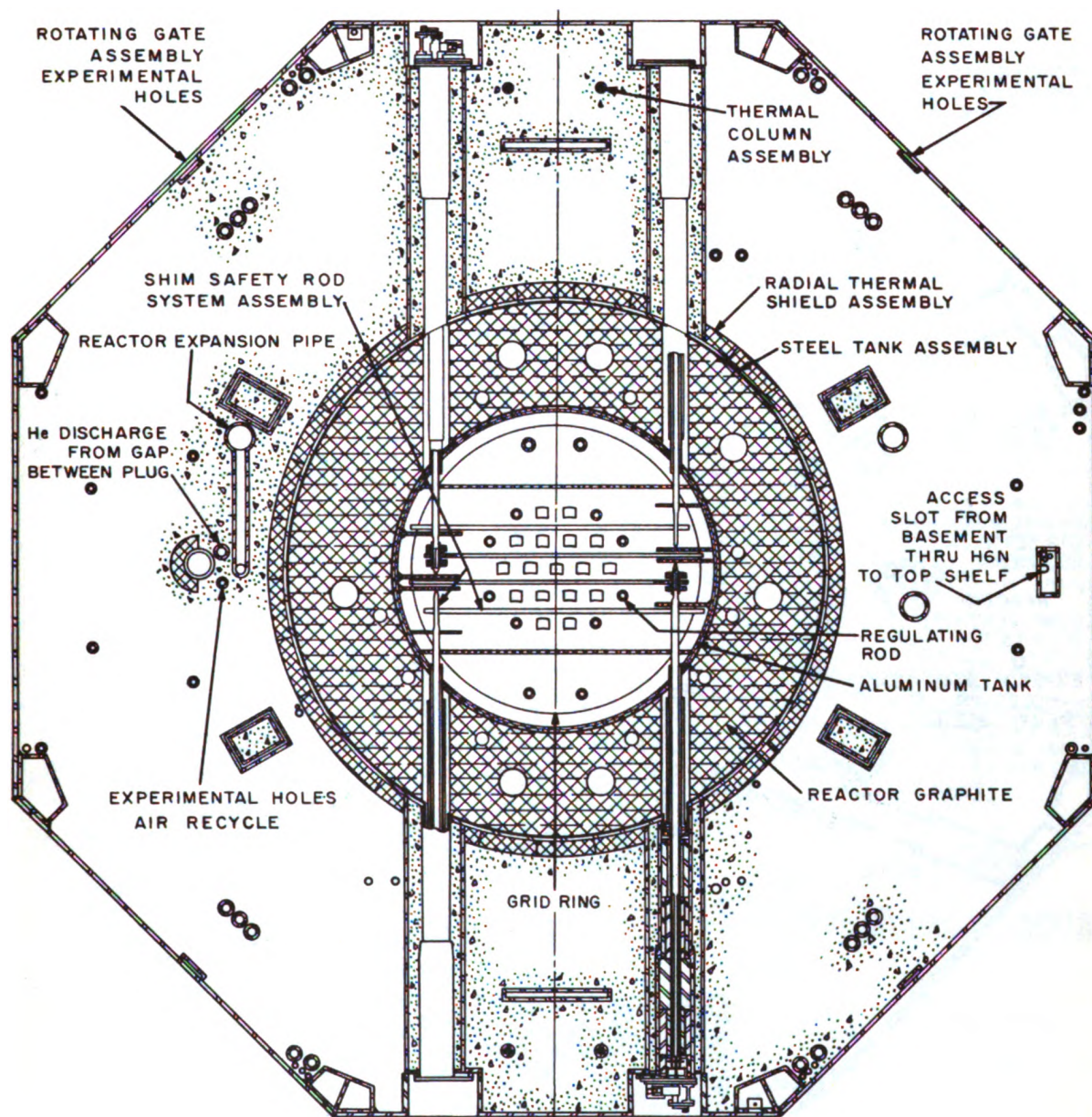


Fig. 5-26 Horizontal section through shim-safety rods.

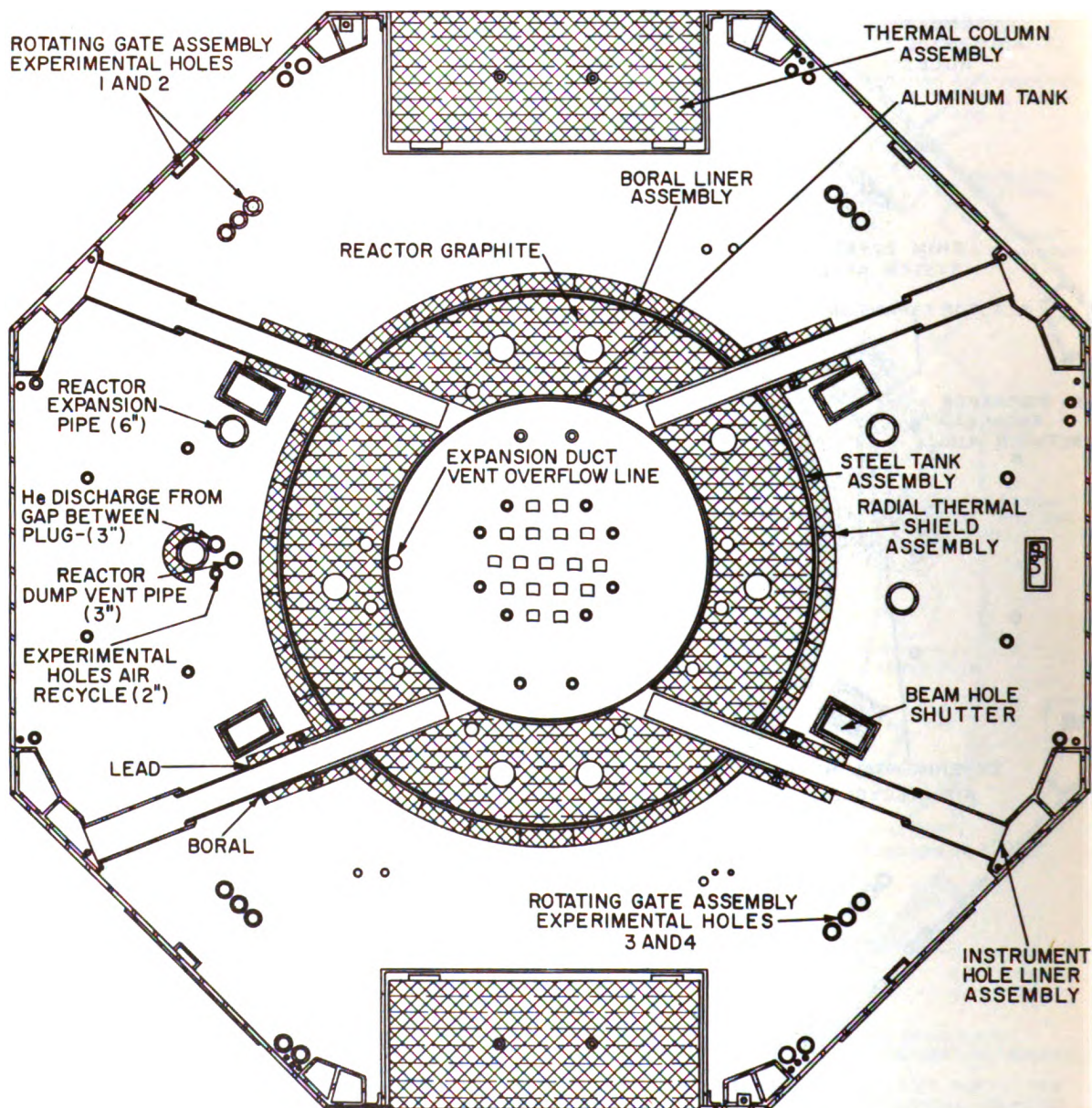
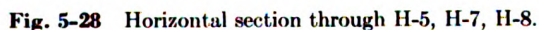


Fig. 5-27 Horizontal section through upper instrument holes.



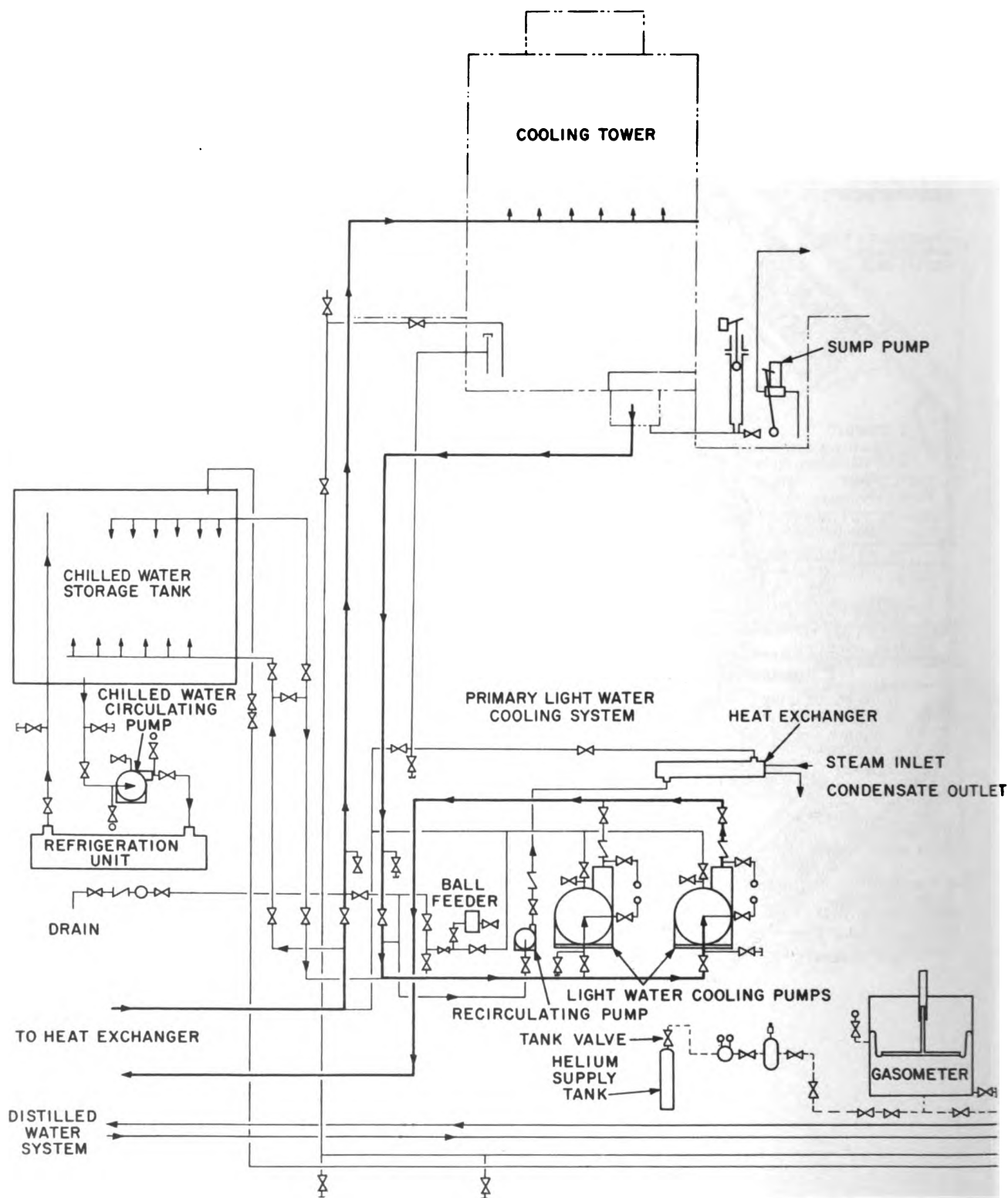


Fig. 5-29A Reactor coolant flow diagram I.

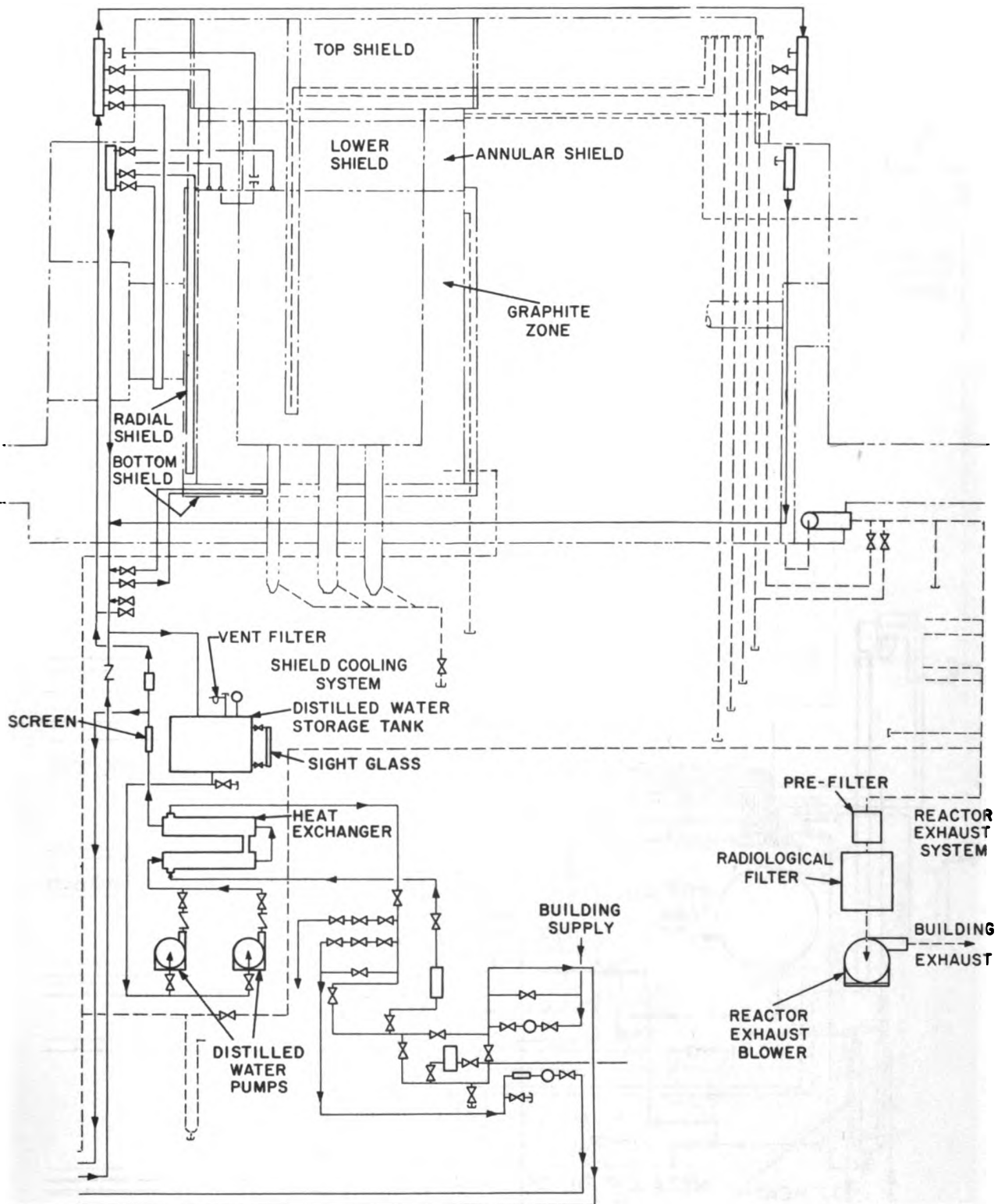


Fig. 5-29B Reactor coolant flow diagram I.

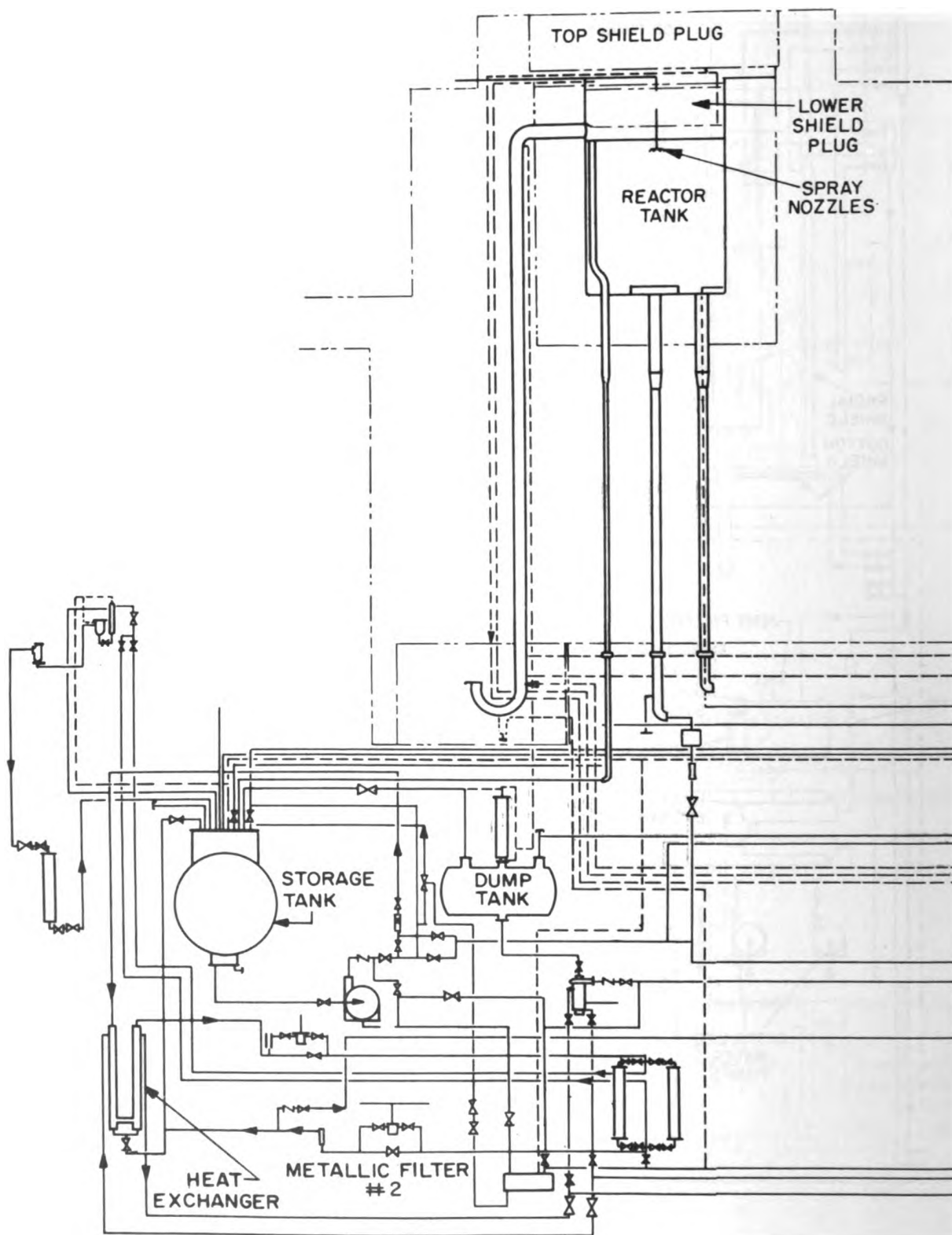


Fig. 5-30A Reactor coolant flow diagram II.

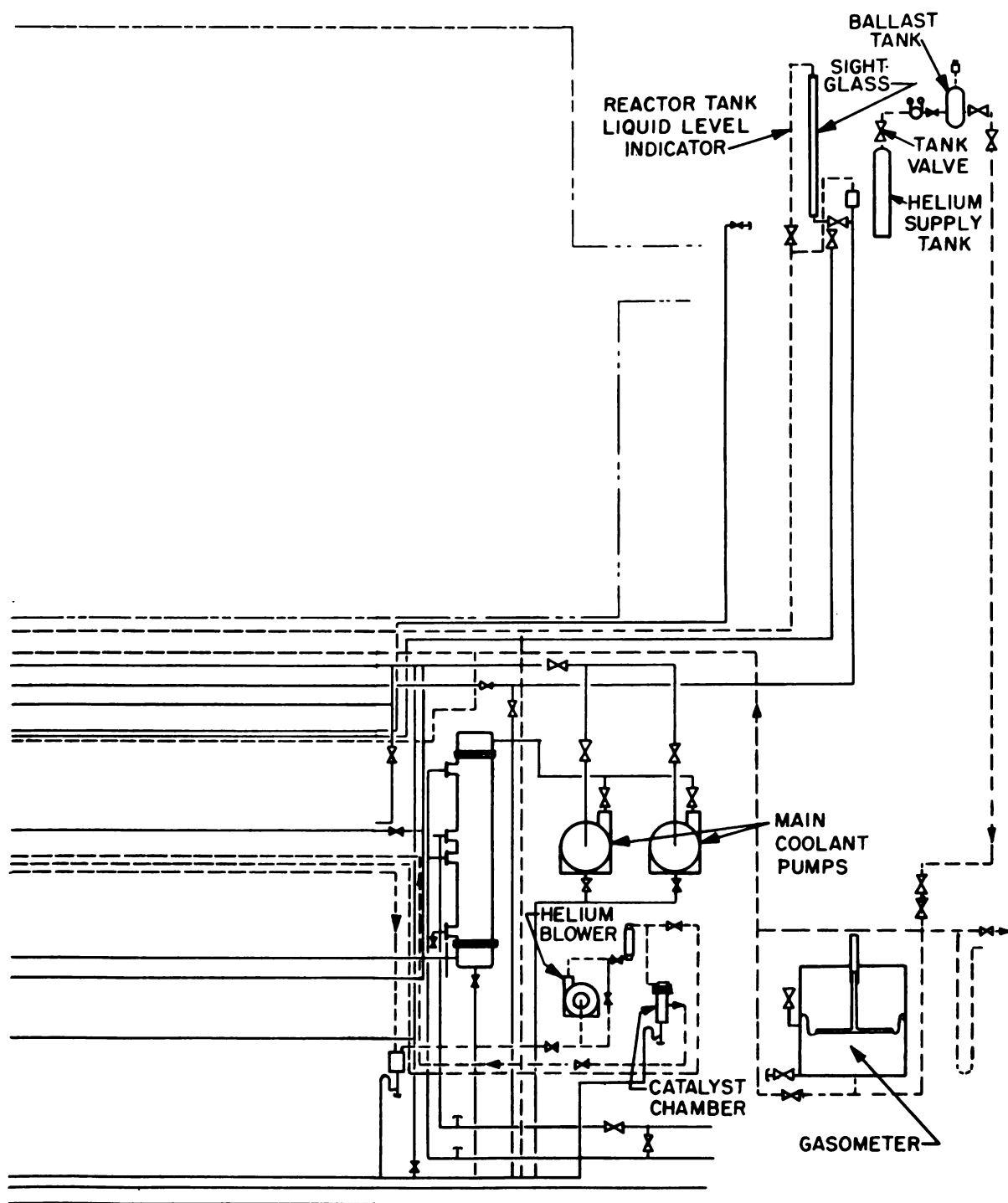


Fig. 5-30B Reactor coolant flow diagram II.

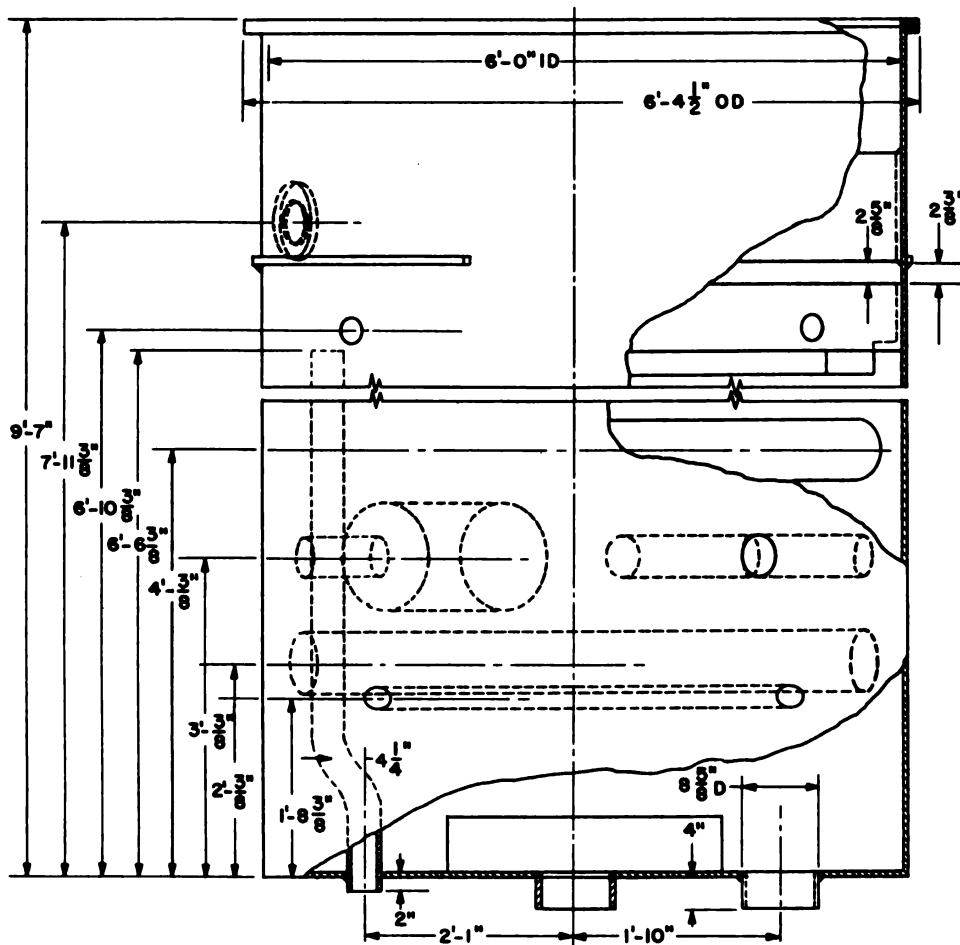
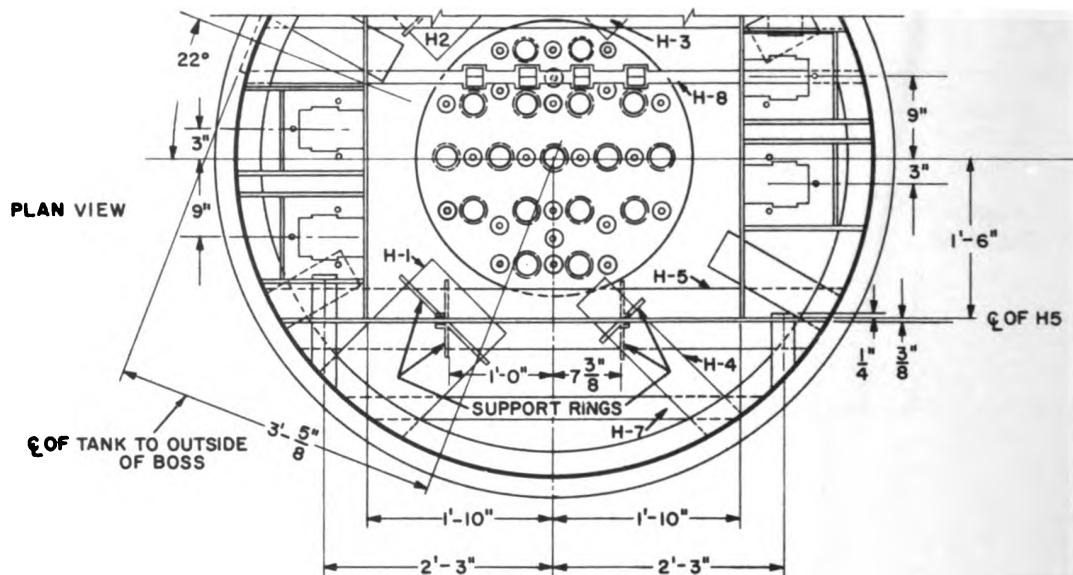


Fig. 5-31 Aluminum tank and accessories.

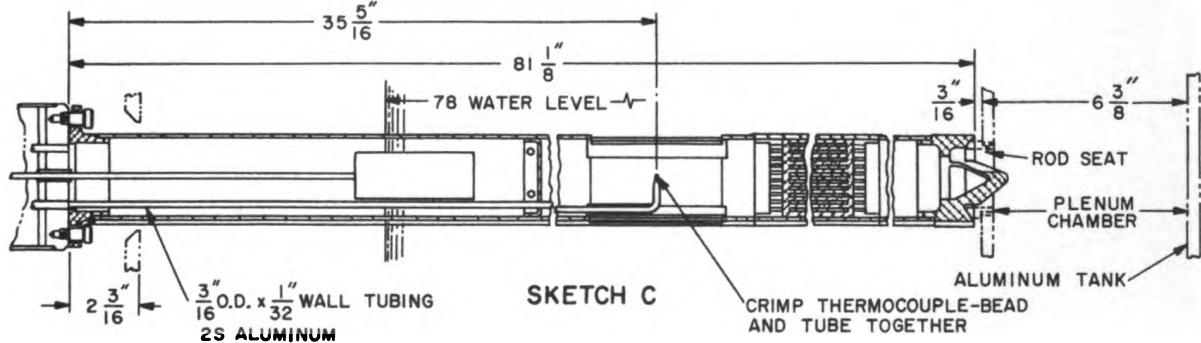
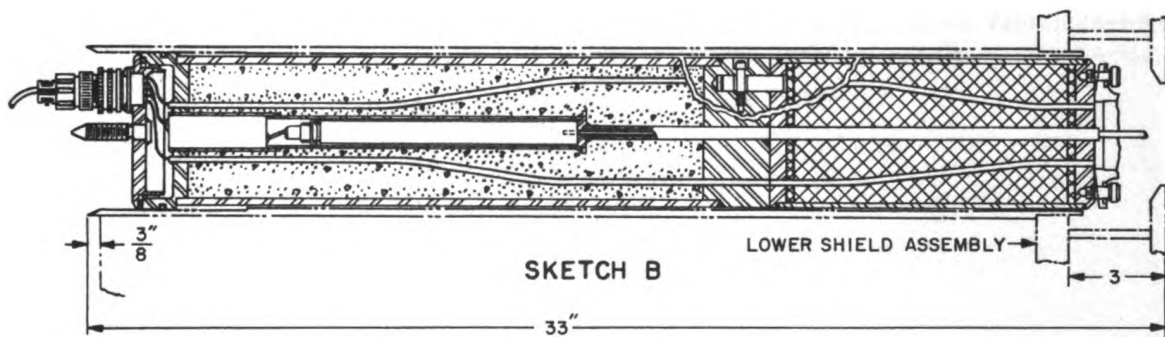
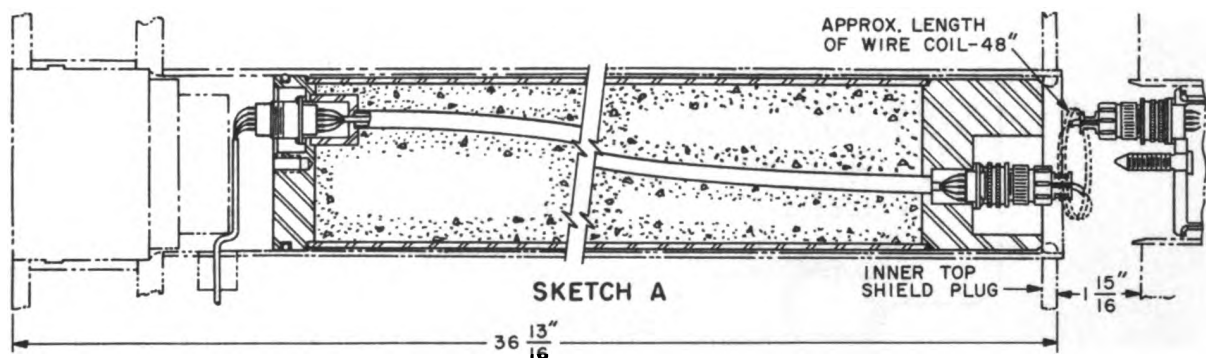
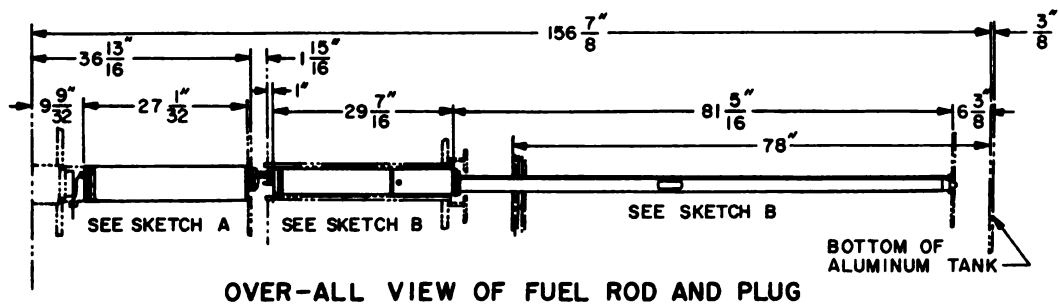


Fig. 5-32 Fuel-rod and plug assembly.

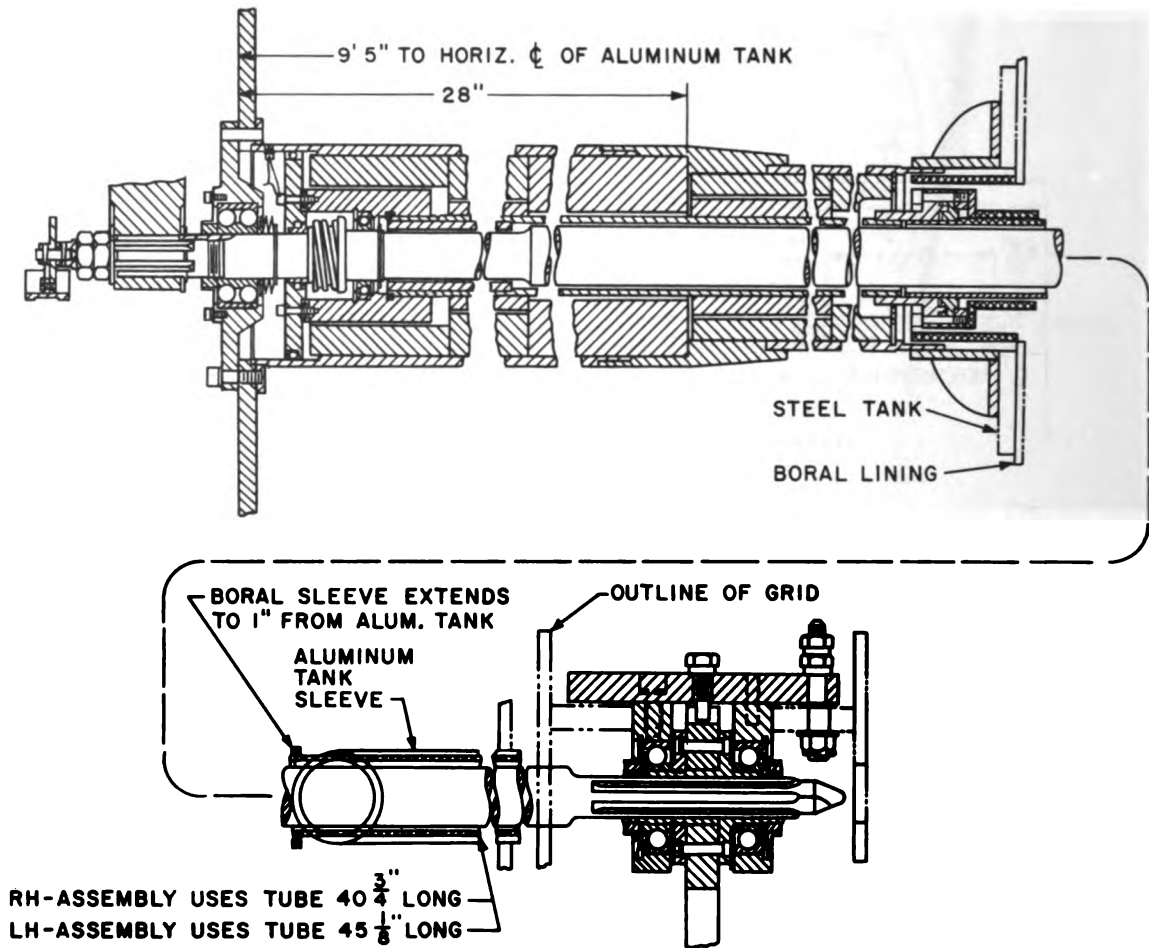


Fig. 5-33 Shim-safety-rod and shaft assembly.

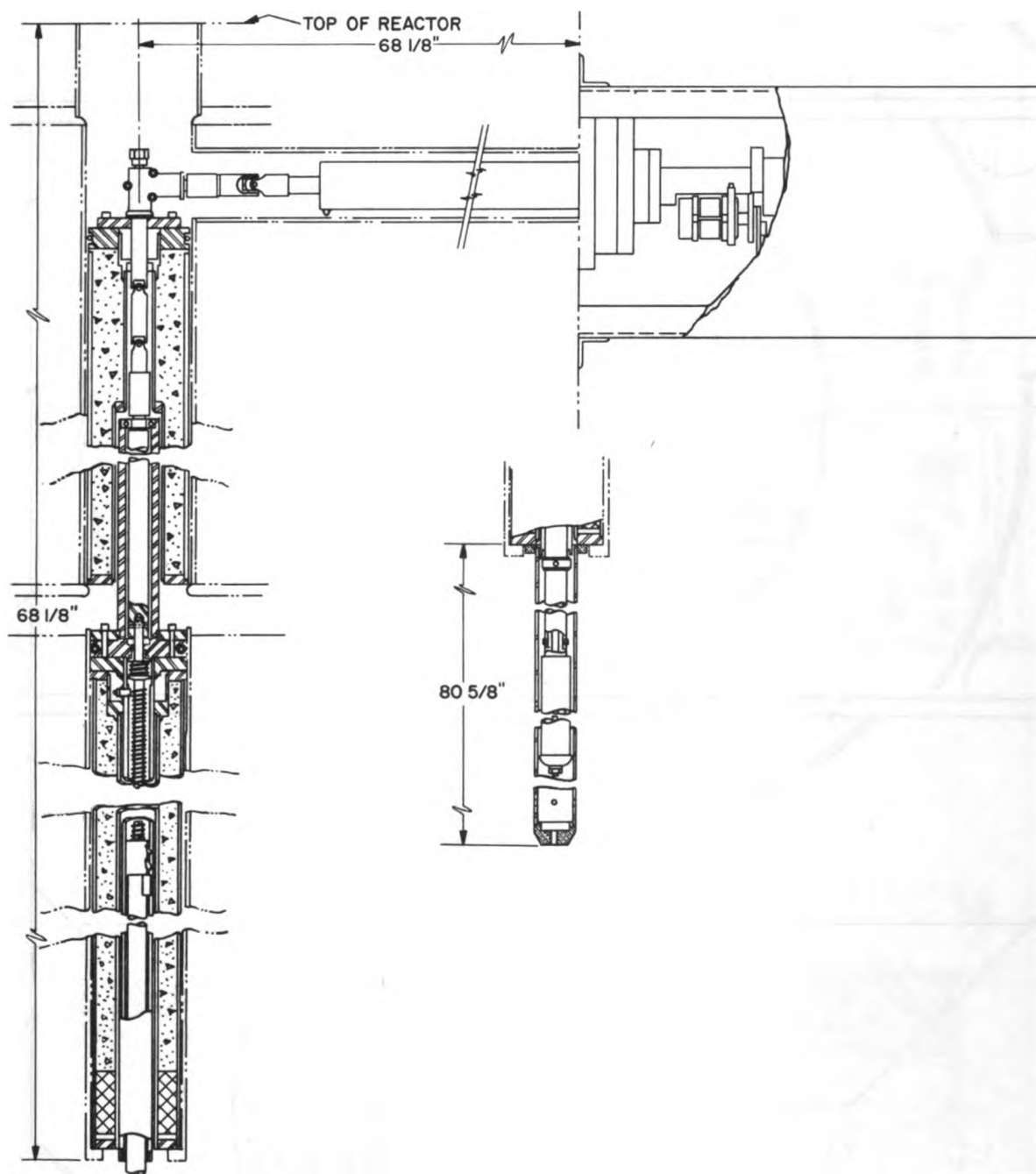


Fig. 5-34 Regulating-rod assembly.

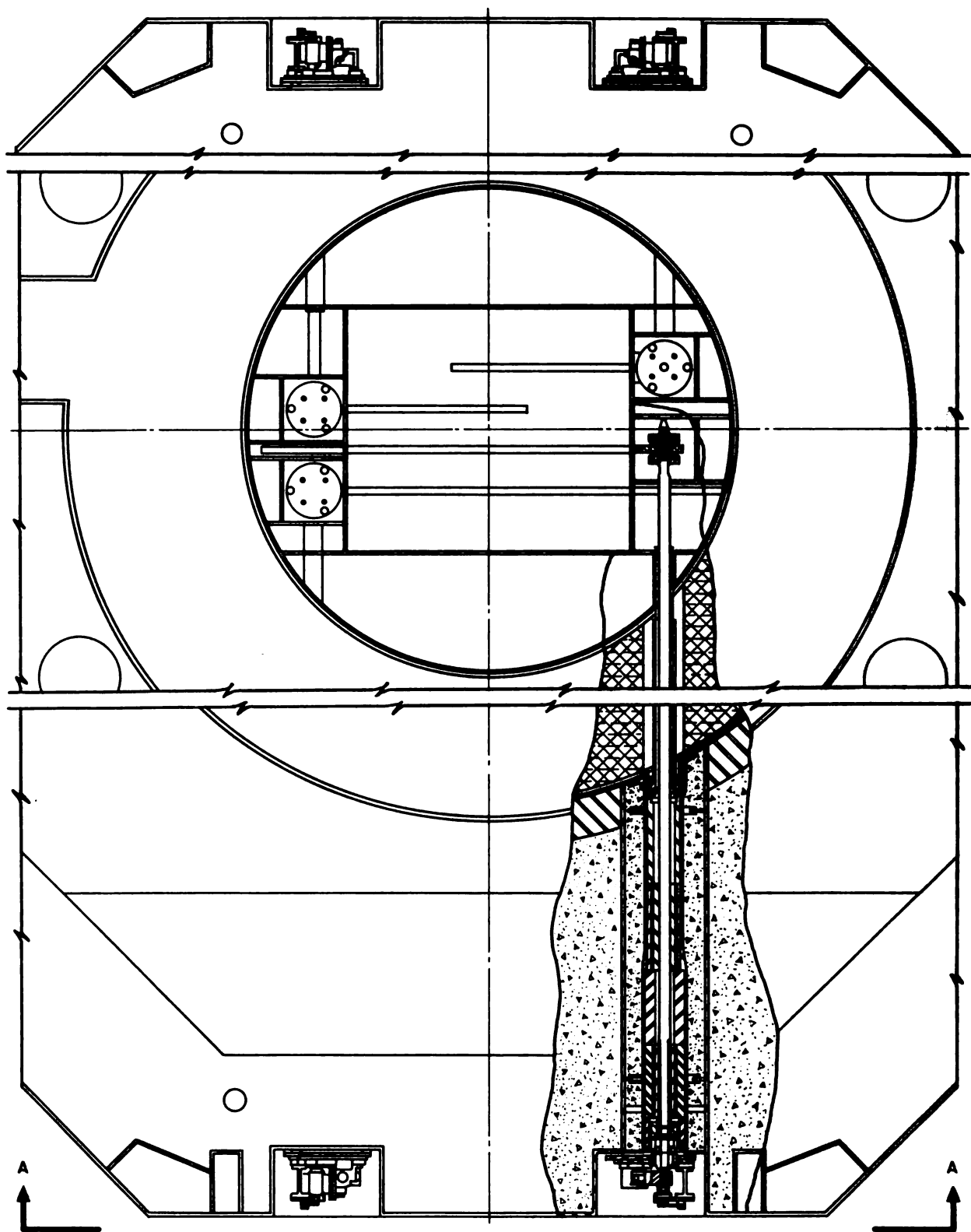


Fig. 5-35A Shim-safety-rod-system assembly.

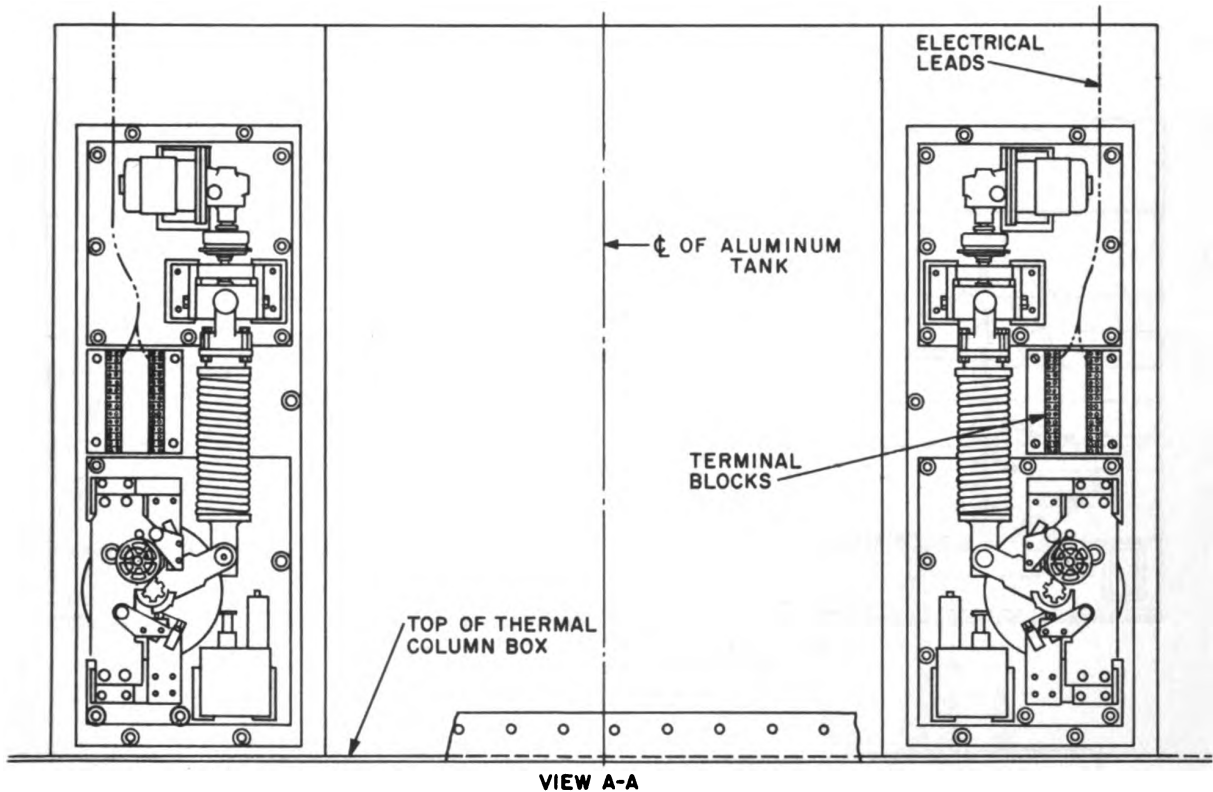


Fig. 5-35B Shim-safety-rod-system assembly.

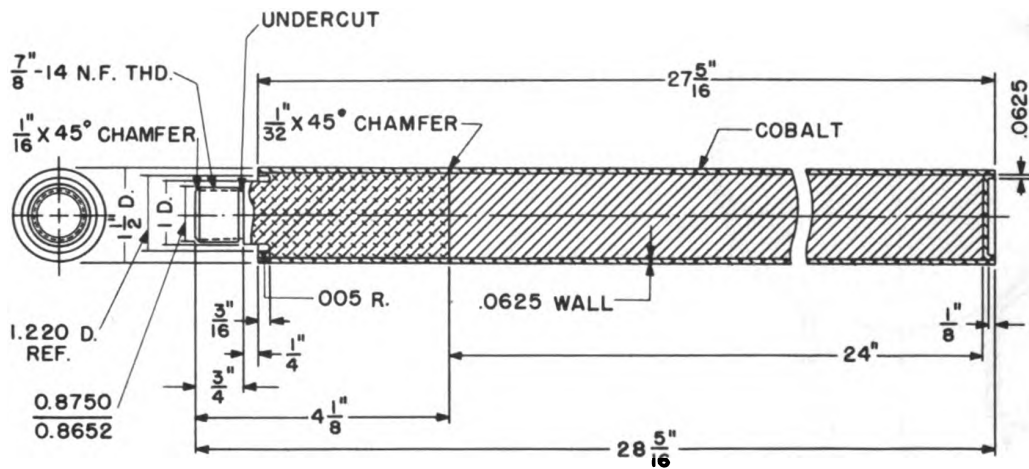
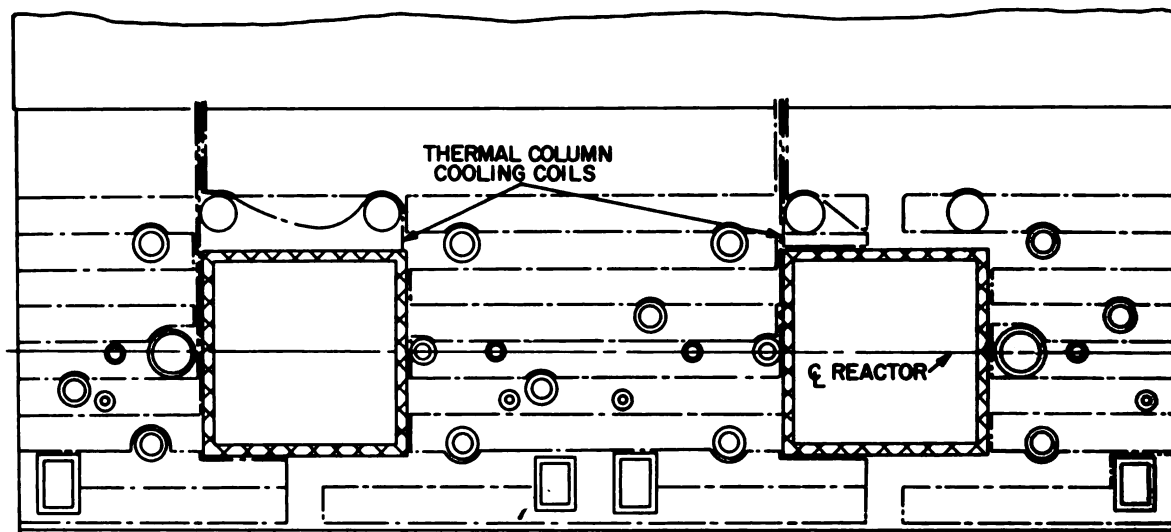


Fig. 5-36 Rod, cobalt-regulating.



FLAT LAYOUT OF TANK SHOWING SCHEMATIC
ARRANGEMENT OF COOLING COILS

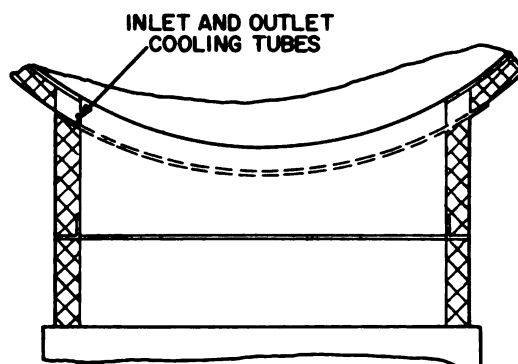
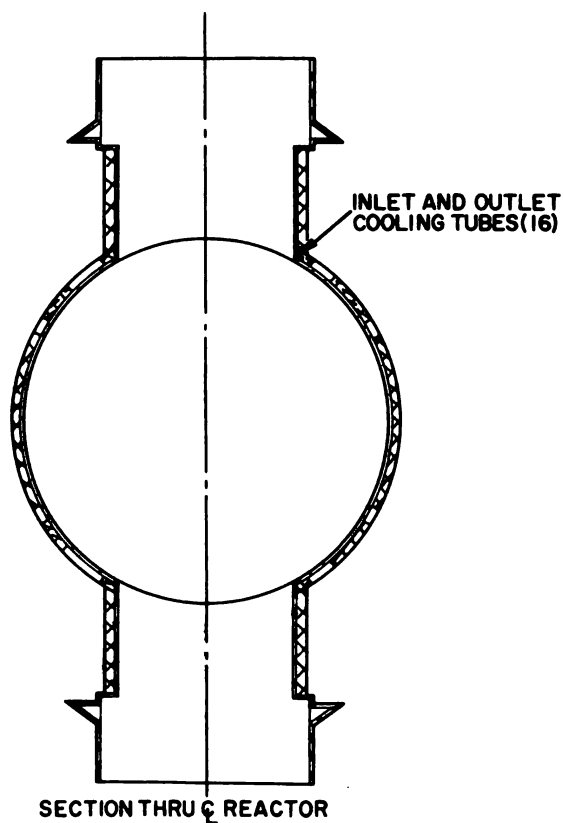


Fig. 5-37 Radial thermal-shield assembly.

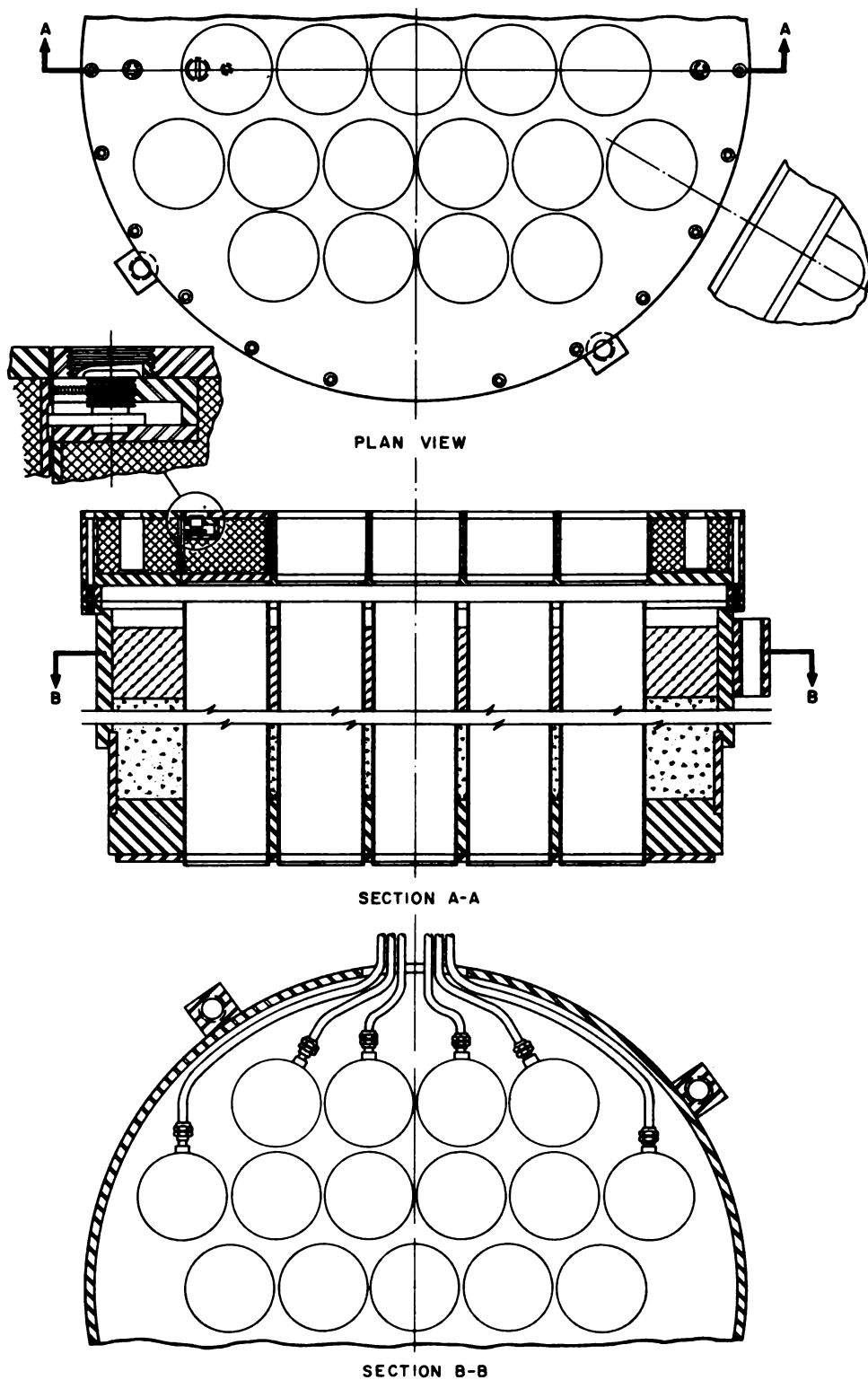
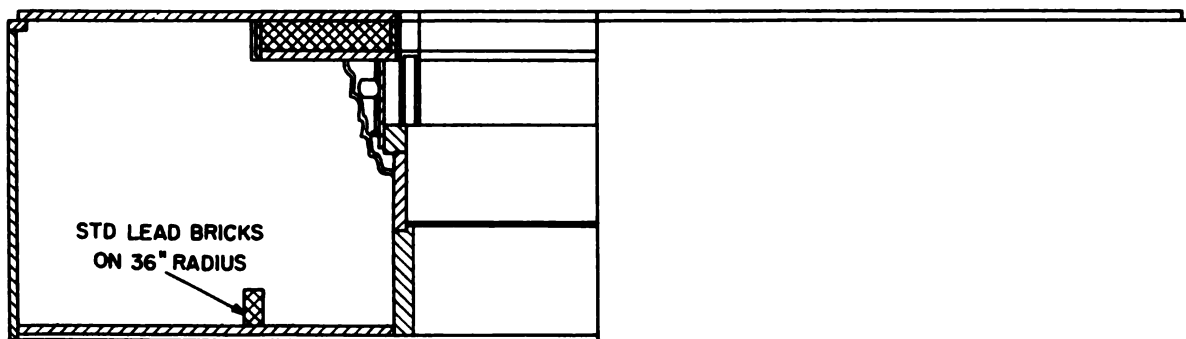
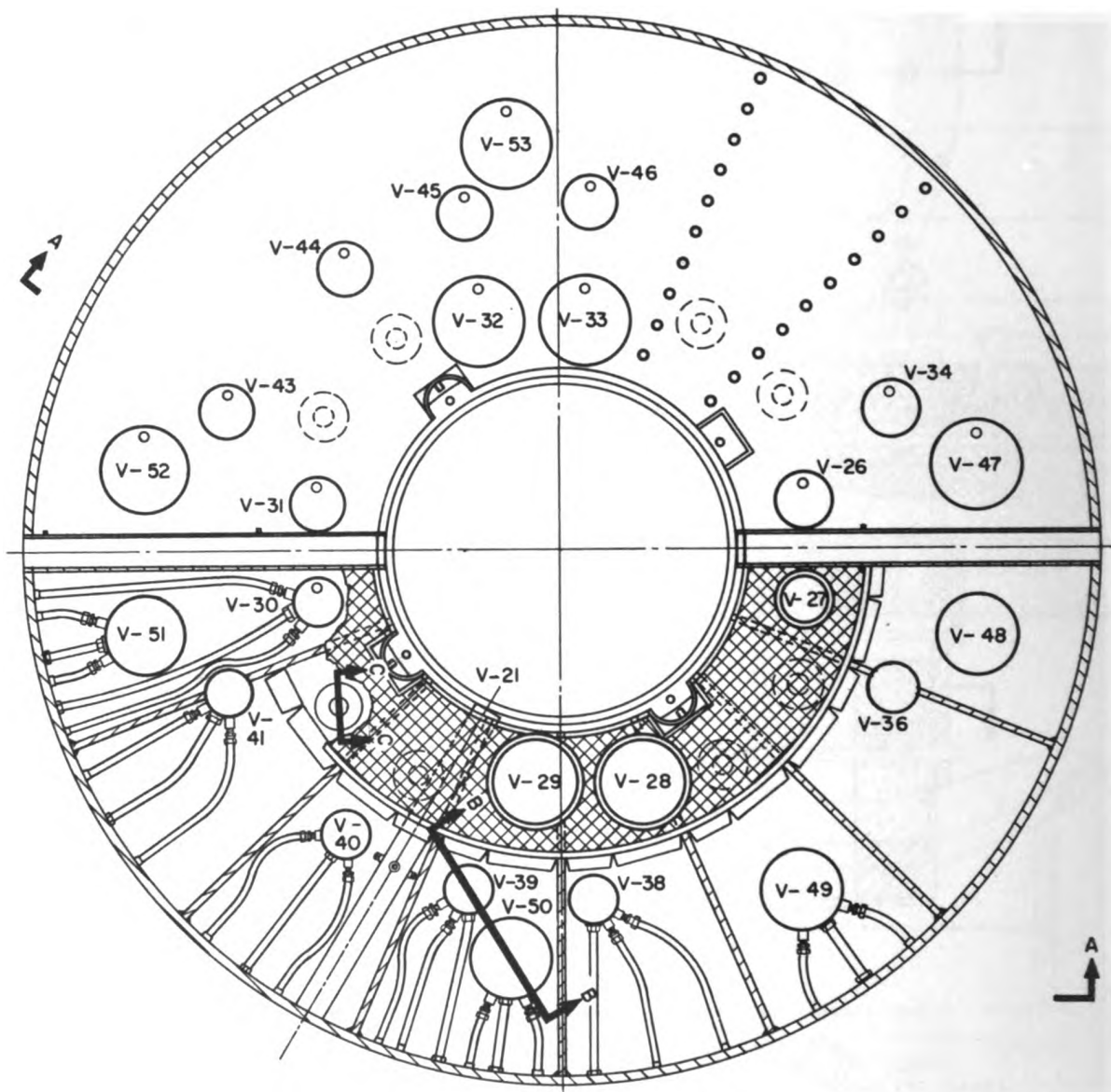
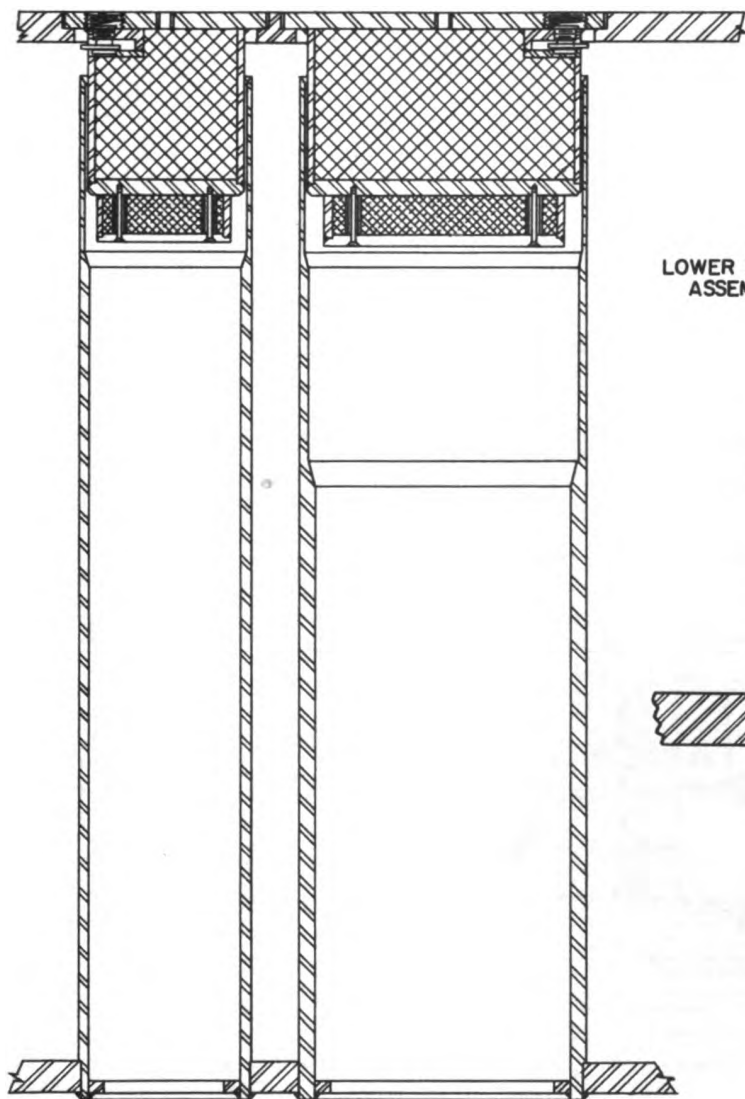


Fig. 5-38 Inner top shield-plug assembly.

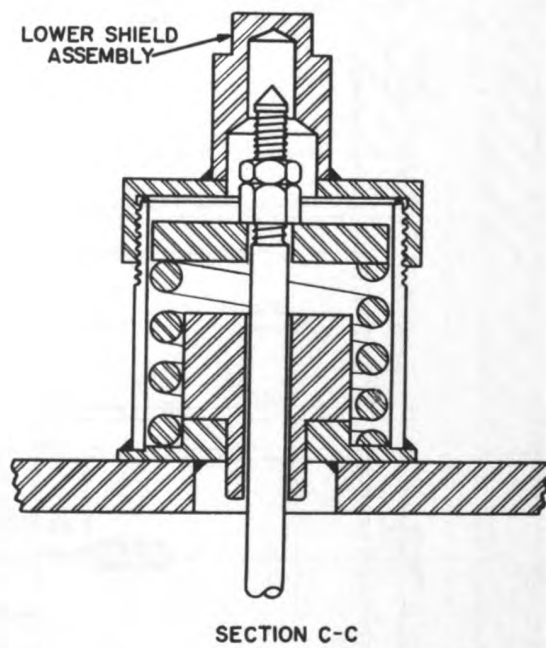


SECTION A-A

Fig. 5-39A Outer top shield-plug assembly.



SECTION B-B



SECTION C-C

Fig. 5-39B Outer top shield-plug assembly.

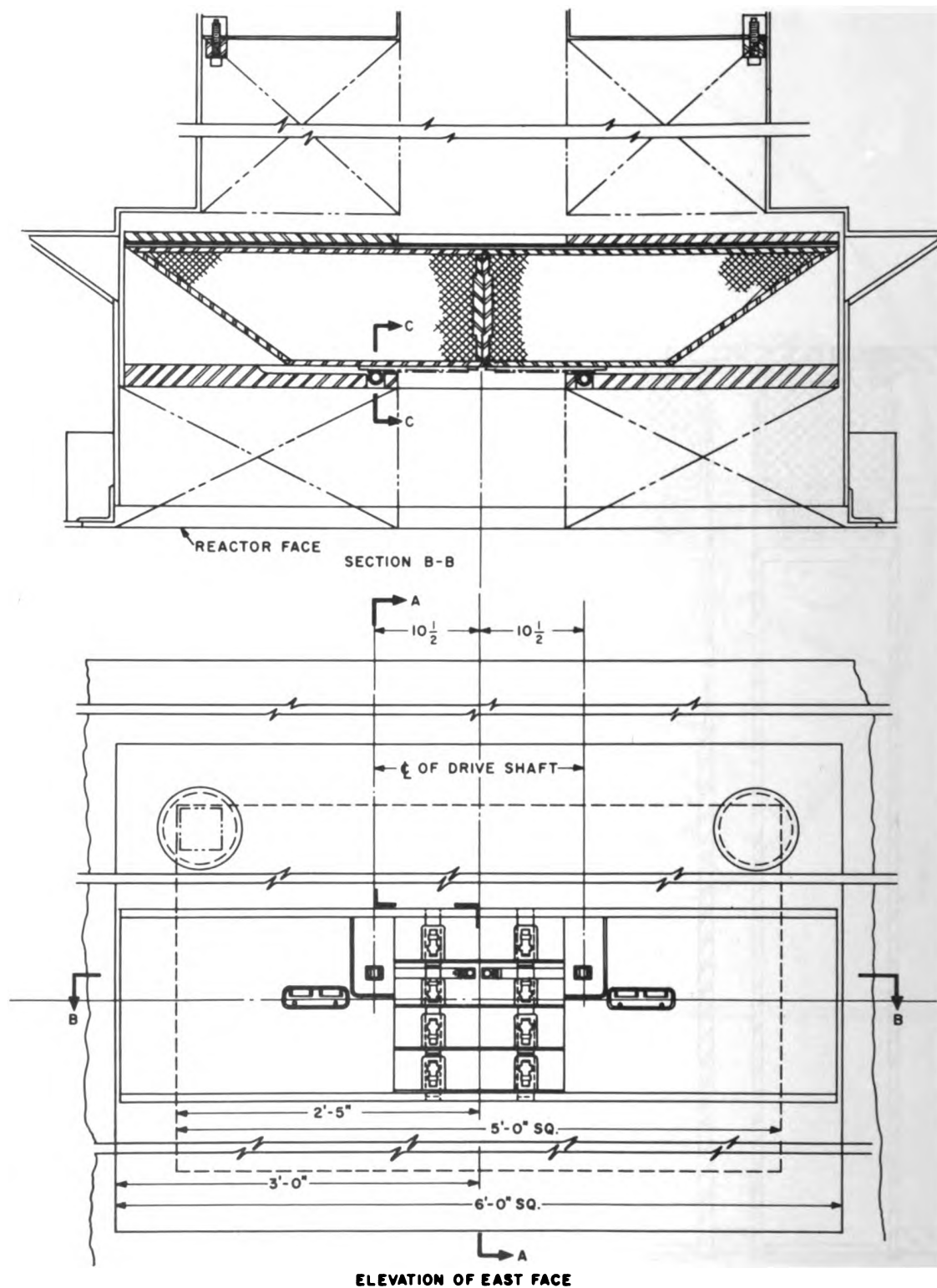
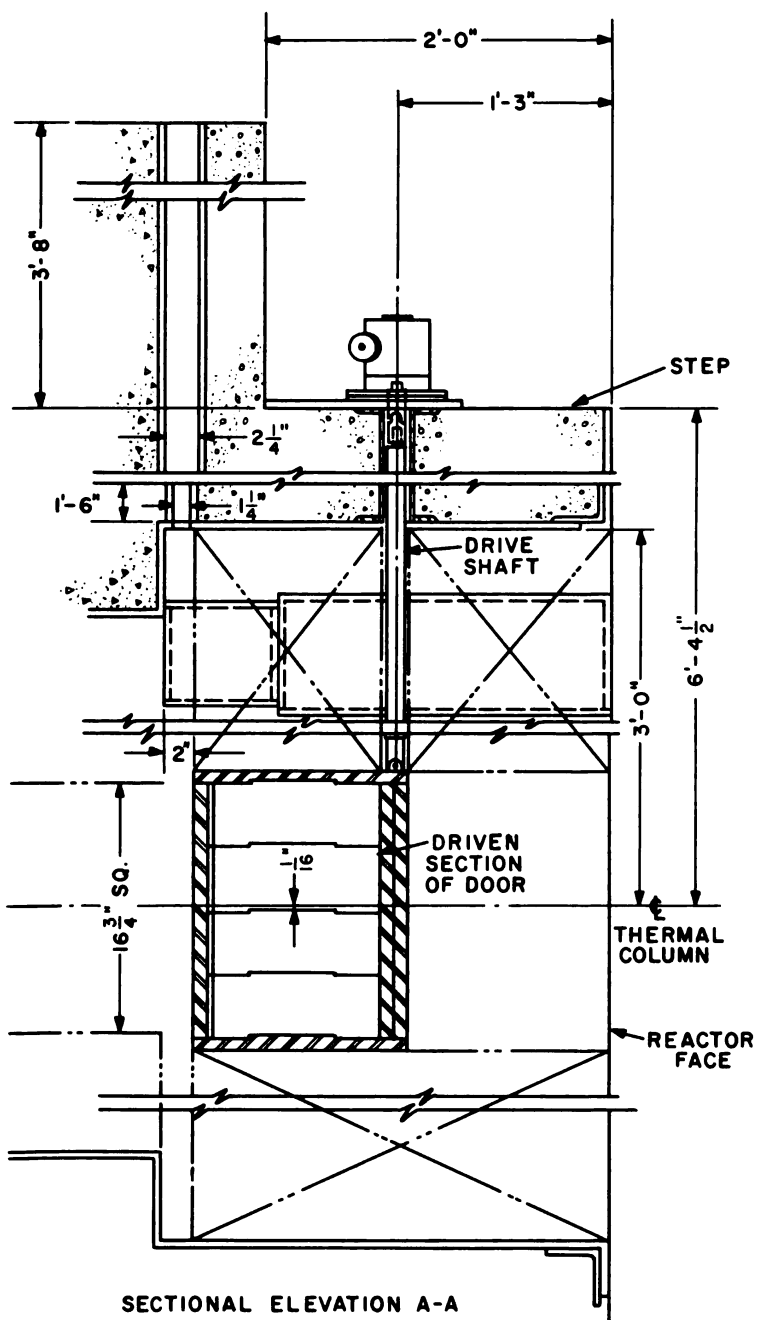
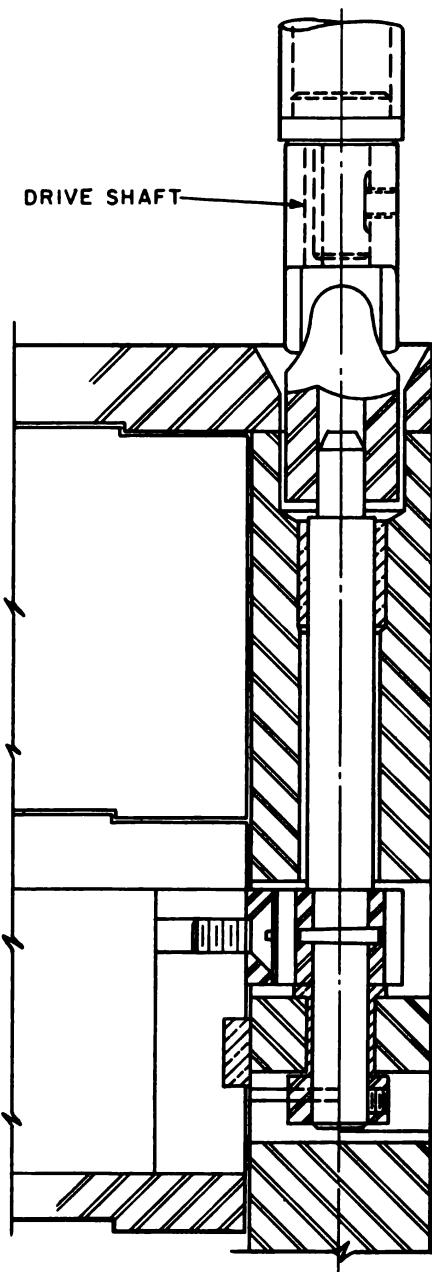


Fig. 5-40 Thermal-column



door assembly.



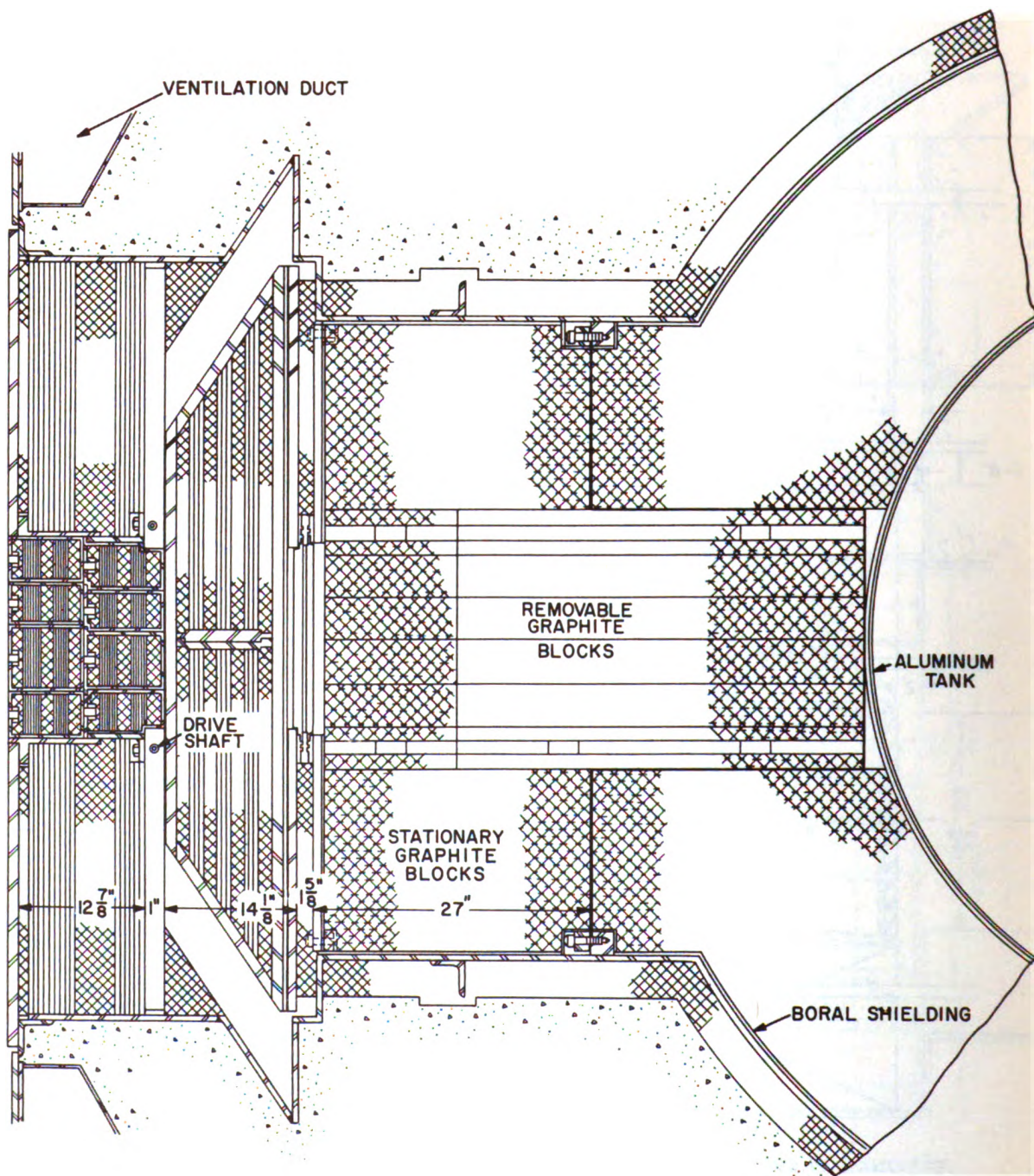


Fig. 5-41 Thermal-column assembly.
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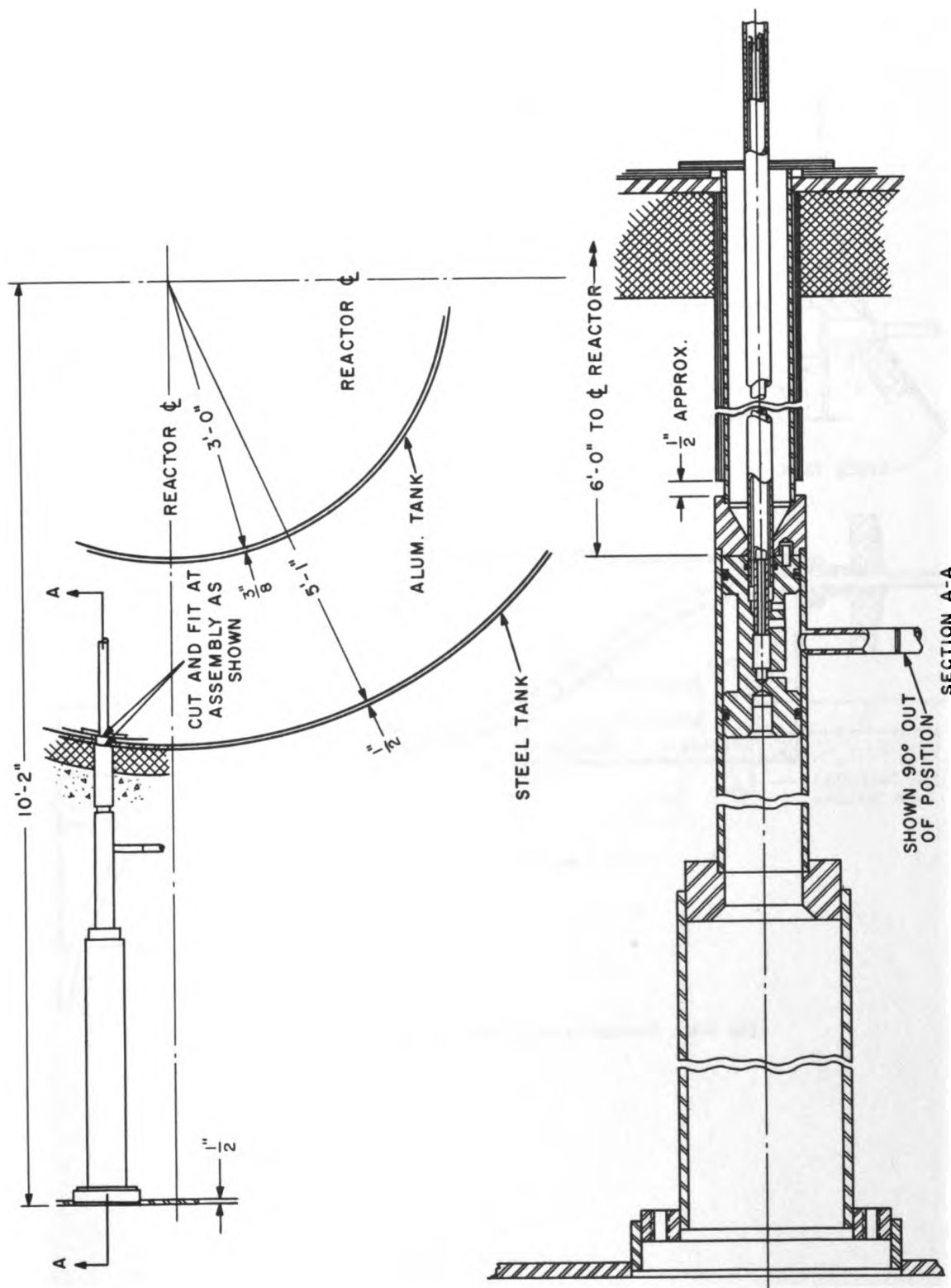


Fig. 5-42 One-inch-pneumatic-tube arrangement, south face.

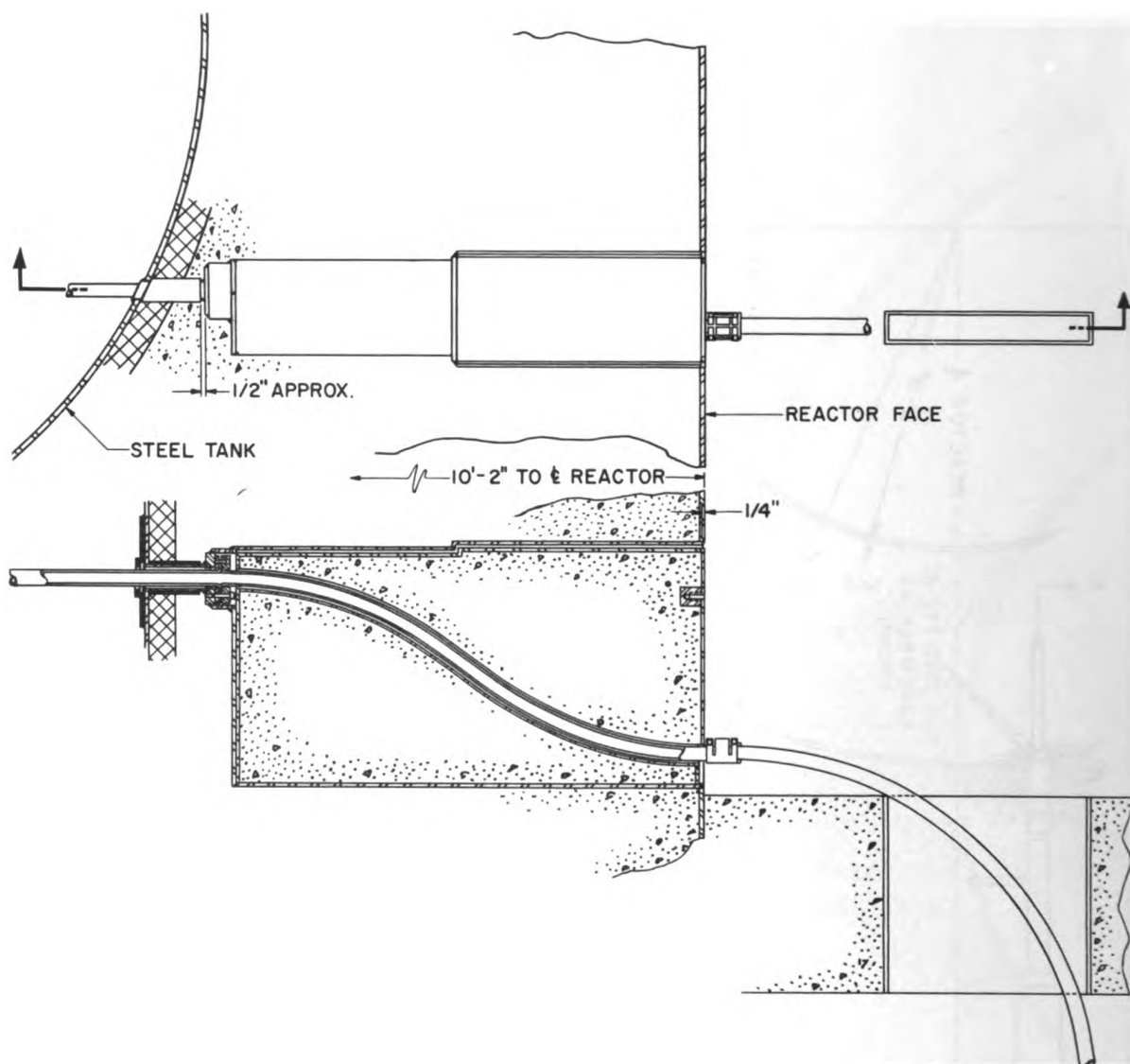


Fig. 5-43 Two-inch-pneumatic-tube arrangement, north face.

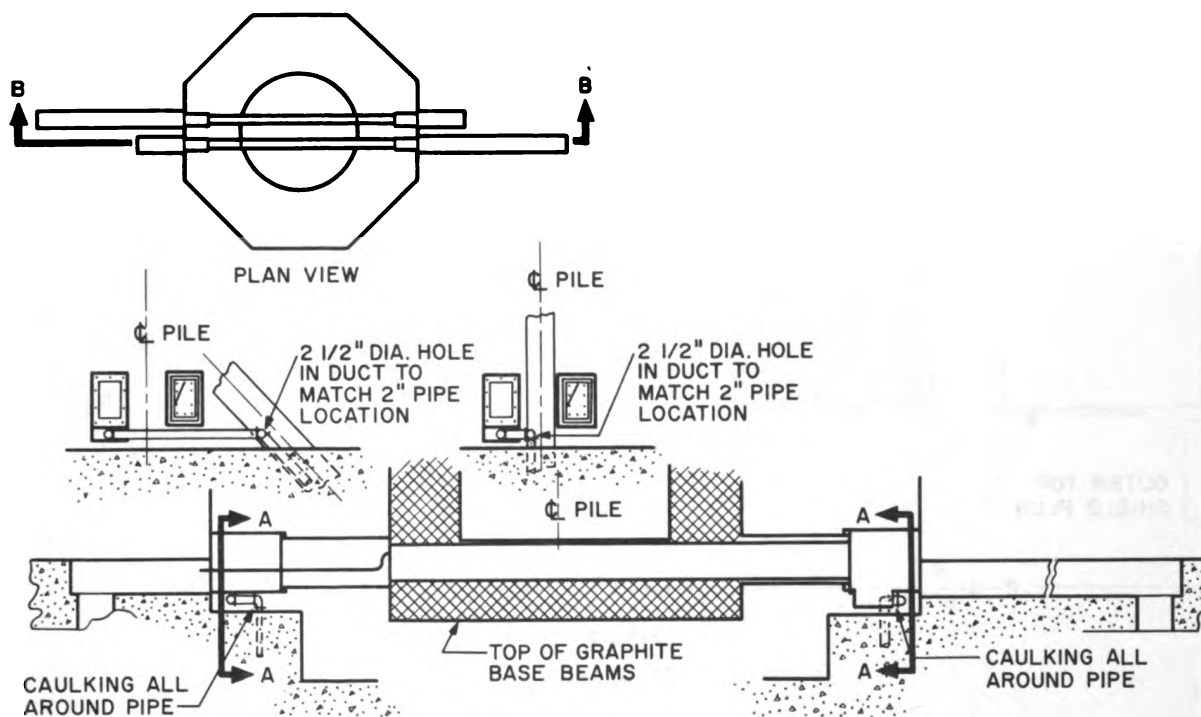


Fig. 5-44 Isotope train-liner assembly.

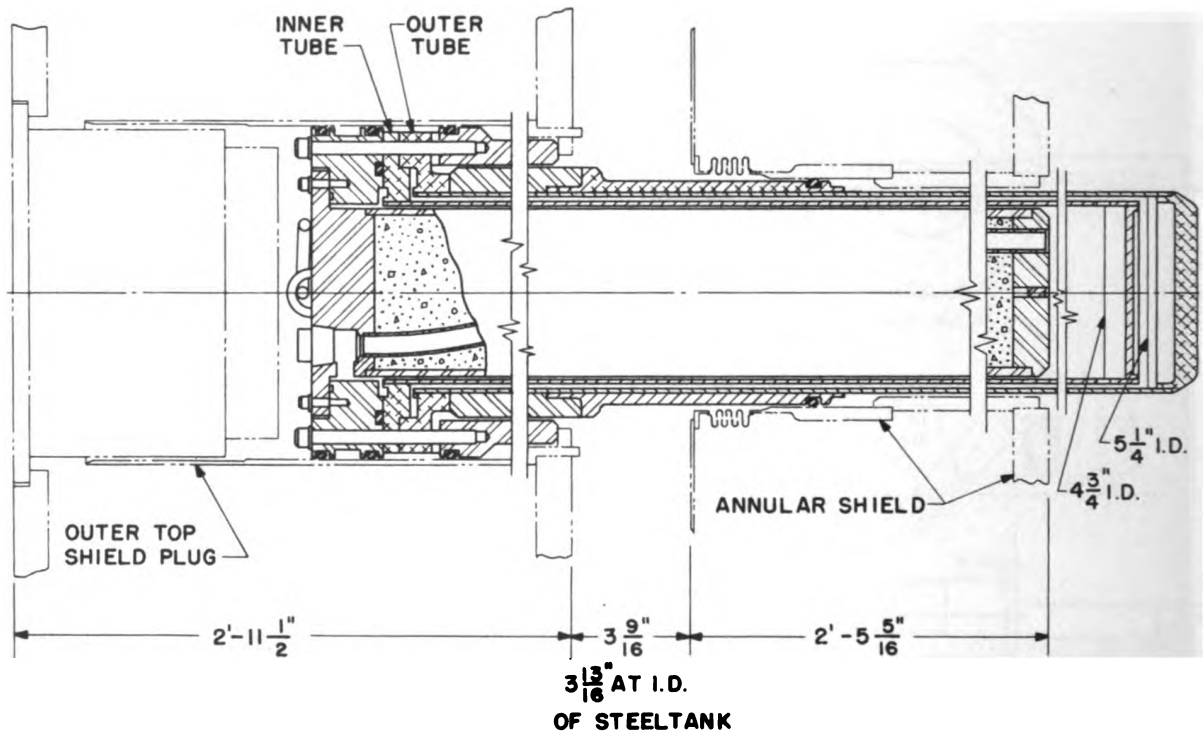


Fig. 5-45 Large graphite-thimble assembly.

CHAPTER 6

Graphite-moderated Reactor

Heterogeneous—Natural Fuel

COMPILED BY BROOKHAVEN NATIONAL LABORATORY
ASSOCIATED UNIVERSITIES, INC.

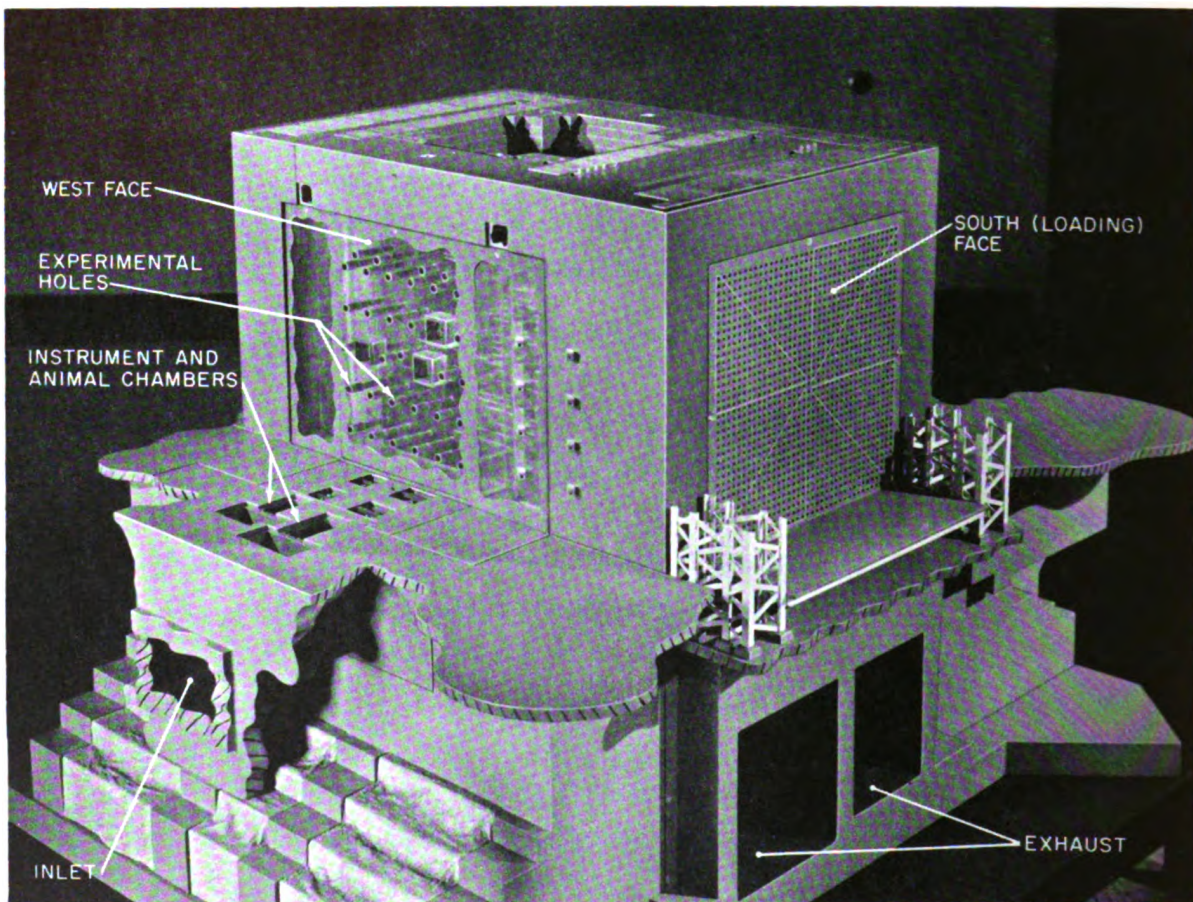


Fig. 6-1 Model of Brookhaven research reactor, a heterogeneous graphite-moderated type, which uses natural uranium fuel. The south face of the reactor is used for loading. Note also the experimental holes and cooling air ducts on the west face.

CHAPTER 6

Graphite-moderated Reactor

Heterogeneous—Natural Fuel

The first large-scale nuclear reactor designed and built specifically to support fundamental research programs was constructed at the Brookhaven National Laboratory (BNL) directly after World War II. The design was based upon the experience gained during the war years with the Oak Ridge and Hanford reactors and was chosen because of the ready availability of its primary materials, reactor-grade graphite and normal uranium metal. It was deemed necessary to design and build the reactor in a minimum period so that it would be available for research as quickly as possible. The project was handled by a private corporation under a contract with the United States Atomic Energy Commission.

The reactor concept was based upon a requirement for large research areas at the reactor, and this necessity influenced the choice of the large-volume graphite-moderated core. A design criterion was a central maximum neutron flux of 5×10^{12} neutrons/cm²/sec, and this dictated a power level of about 30,000 kw, a much higher level than that of any gas-cooled reactor existing at that time. About 1,500,000 lb/hr of air is required for the heat removal. In order to satisfy the requirements for a great variety of research facilities, the inherent complexity of the graphite structure, due to fuel channels and control rods, was multiplied manifold. A high-density iron-loaded concrete was developed as a shielding material in order to achieve adequate attenuation in a minimal distance of 5 ft, thus ensuring low backgrounds for research purposes. The horizontal control rods were positioned along the diagonals of the square section of the graphite structure to free all faces of the structure for research areas.

A great variety of research facilities were an integral part of the original plans. A pattern of 30 experimental beam holes was placed in the horizontal direction, perpendicular to the fuel channels, and continuous through the graphite structure and the two shielding walls. A machine for the irradiation of small target items was built to handle great numbers of such items in and out of the reactor without the need of reactor shutdowns. There is also a system of pneumatically operated tubes for short-period irradiations. Other irradiation facilities permit irradiations under controlled temperature conditions from -190 to 500°C .

The mode of operation, suited to the large research program, is based on the following precepts:

1. Continuous around-the-clock operation at maximum power levels.
2. Brief shutdowns every 10 days (alternate Tuesday and Friday evenings) to allow for changes in experimental equipment. These shutdowns are also used for replacement of fuel rods and maintenance work.
3. An operating philosophy which emphasizes that the reactor exists to support research and not to produce heat.
4. A minimum of red tape.

Most of the beam holes are used by various research programs for an extended period. Thus, once the equipment is set up, the program continues to develop new data for months and years. There are some 30 separate experiments at the BNL reactor which have been in progress for more than a year. The rest of the beam holes are used intermittently for shorter-term experiments, which may last only weeks or months. This is also true of the irradiation facilities.

Flexibility is the most important aspect of the facility, where research is done by academic institutions, by industry, and by hospitals. There is no end to the variety of problems proposed by research people.

The BNL reactor is a large research reactor of the graphite-moderated type, which utilizes natural uranium fuel in a heterogeneous arrangement. A detailed description of the BNL reactor

exposing materials to neutron bombardment, for experimenting with beams of radiation, and for producing radioisotopes of high specific activity. Figure 6-2 shows a night view of the buildings housing the reactor; Fig. 6-3 is a plan view of the reactor building.

Reactor. The reactor proper consists of a 25-ft graphite cube separated into two halves by a gap

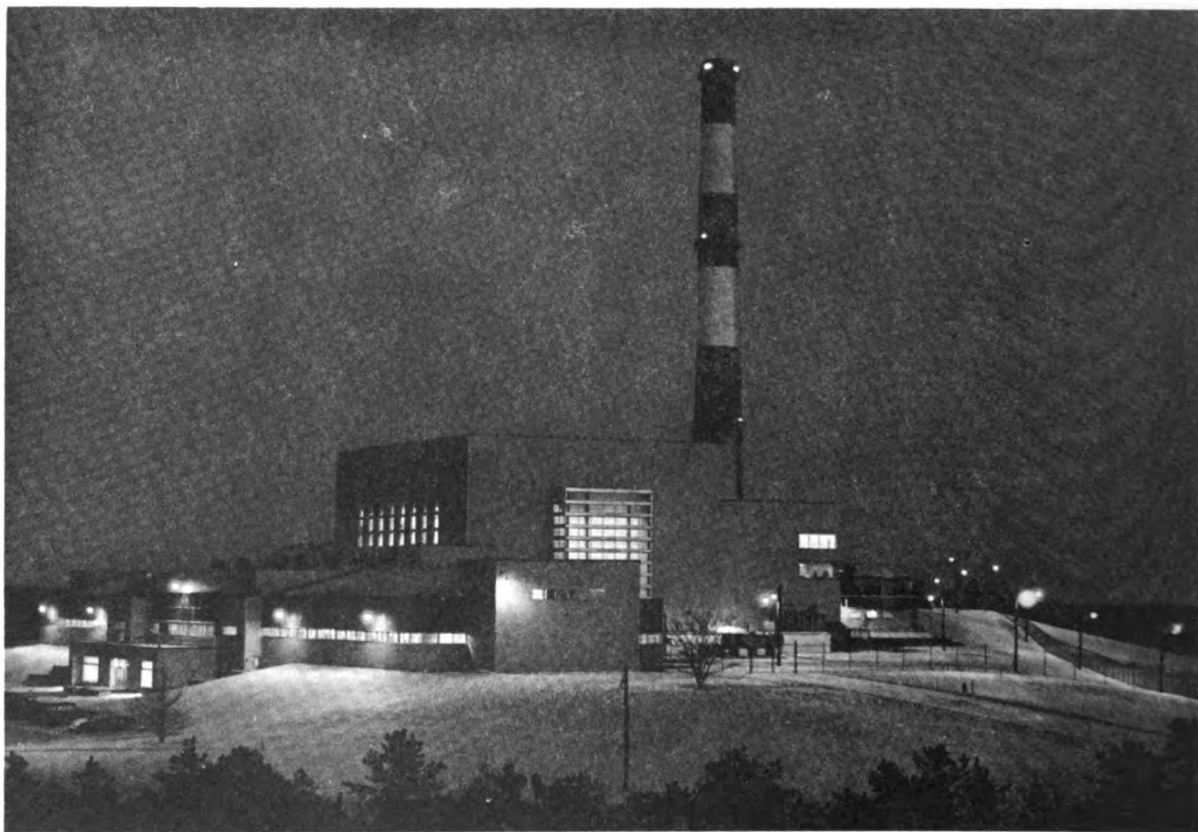


Fig. 6-2 Night view of Brookhaven research reactor buildings.

follows, including data and illustrations on core design, shielding, cooling, controls and instrumentation, and the various experimental facilities. A number of engineering drawings of the major systems are included in the Appendix.

GENERAL DESCRIPTION

This reactor is of the forced-air-cooled natural-uranium graphite-moderated thermal type and provides a maximum flux of 6×10^{12} neutrons/cm²/sec; it is designed to produce heat at the rate of 30,000 kw. There are special provisions for

of 2.75 in. in the vertical east-west plane. Sixty thousand graphite bricks with a total weight of 700 tons make up the core, which is penetrated in the north-south direction by 1368 round, parallel fuel channels spaced 8 in. between centers. A sufficient number of the channels are filled with uranium slugs 11 ft long and encased in 11-ft-long finned aluminum cartridges to bring the activity of the reactor to the desired level. These fuel cartridges are filled with helium under slight pressure. This provides an inert atmosphere for the uranium and also facilitates the detection of cartridge failures by means of pressure changes. Fig-

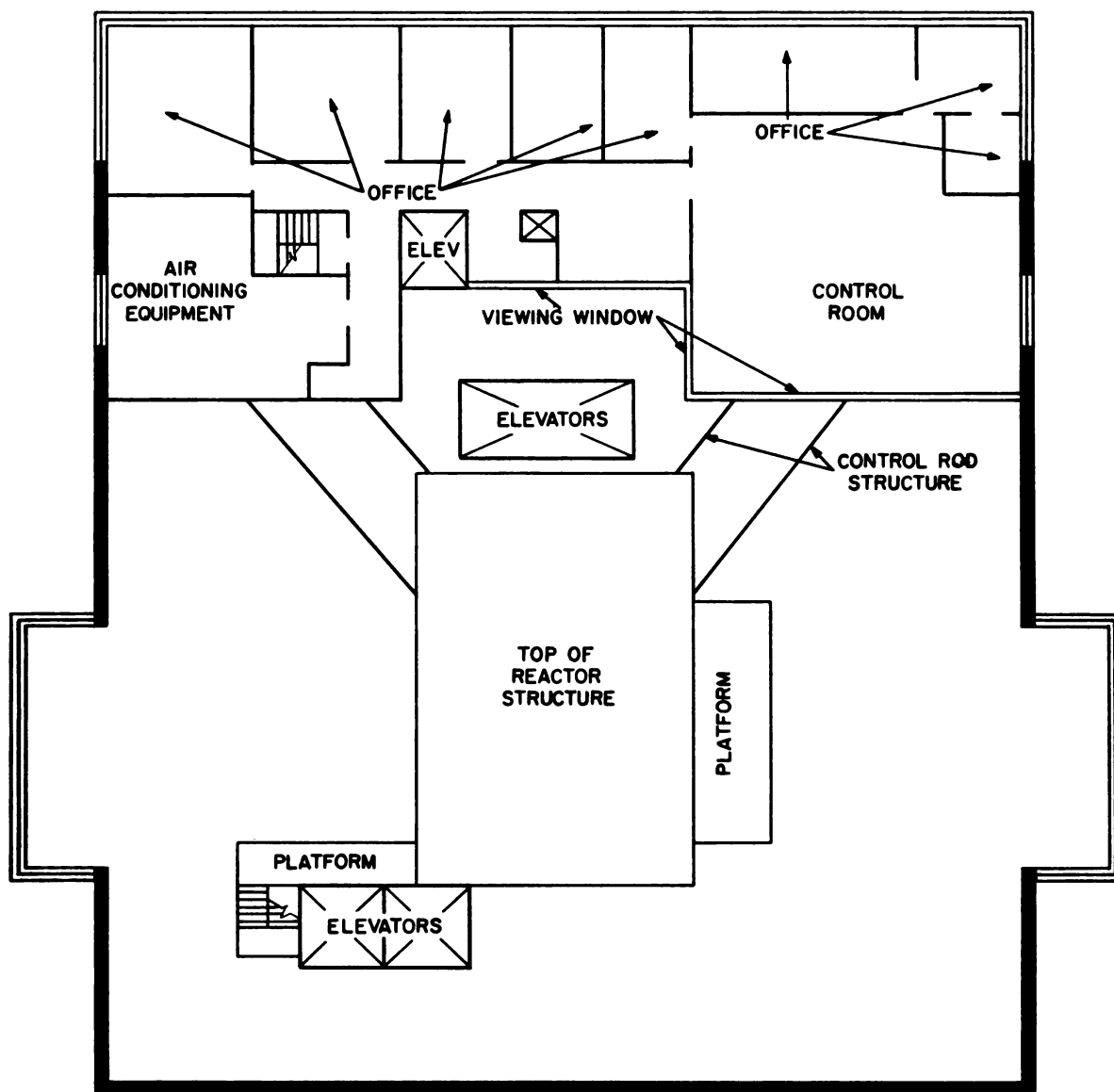


Fig. 6-3 Plan view of reactor building showing location of reactor.

ure 6-4 is a top view of the reactor model. The black area in the center represents the graphite; the white rods are the fuel elements. The shielding blocks on top are removable for experimental use.

Cooling. The reactor is cooled by a stream of air which is drawn into the gap and flows out through the channels in both directions, removing heat from the aluminum cartridges and the graphite matrix.

The graphite cube and air chambers are en-

closed in a concrete shield 5 ft thick (provided with plenum chambers), thus bringing the overall width to 38 ft and the length to 55 ft.

Five 1500-hp fans draw air through the reactor at the rate of 750 tons/hr and exhaust it through a 320-ft stack. Air speeds vary from 40 mph in the two main ducts to hundreds of miles per hour in the fuel channels. The air is filtered both before and after leaving the reactor. Before reaching the fans it is also cooled by water to reduce its volume and thereby cut down the required

power; the lowered air temperature also prevents dehydration of the concrete in the ducts.

Control Rods and Safety Systems. The control rods, made of steel alloy containing a small percentage of boron, run into the reactor diagonally from the charging face so that the sides

can be released manually from the control room into vertical aluminum tubes and (2) liquid trichlorobenzene which can be released to flood pneumatic tubes running through the reactor. The natural draft of the stack is sufficient to cool the reactor after an emergency shutdown caused by loss of forced air-cooling. There are storage bat-

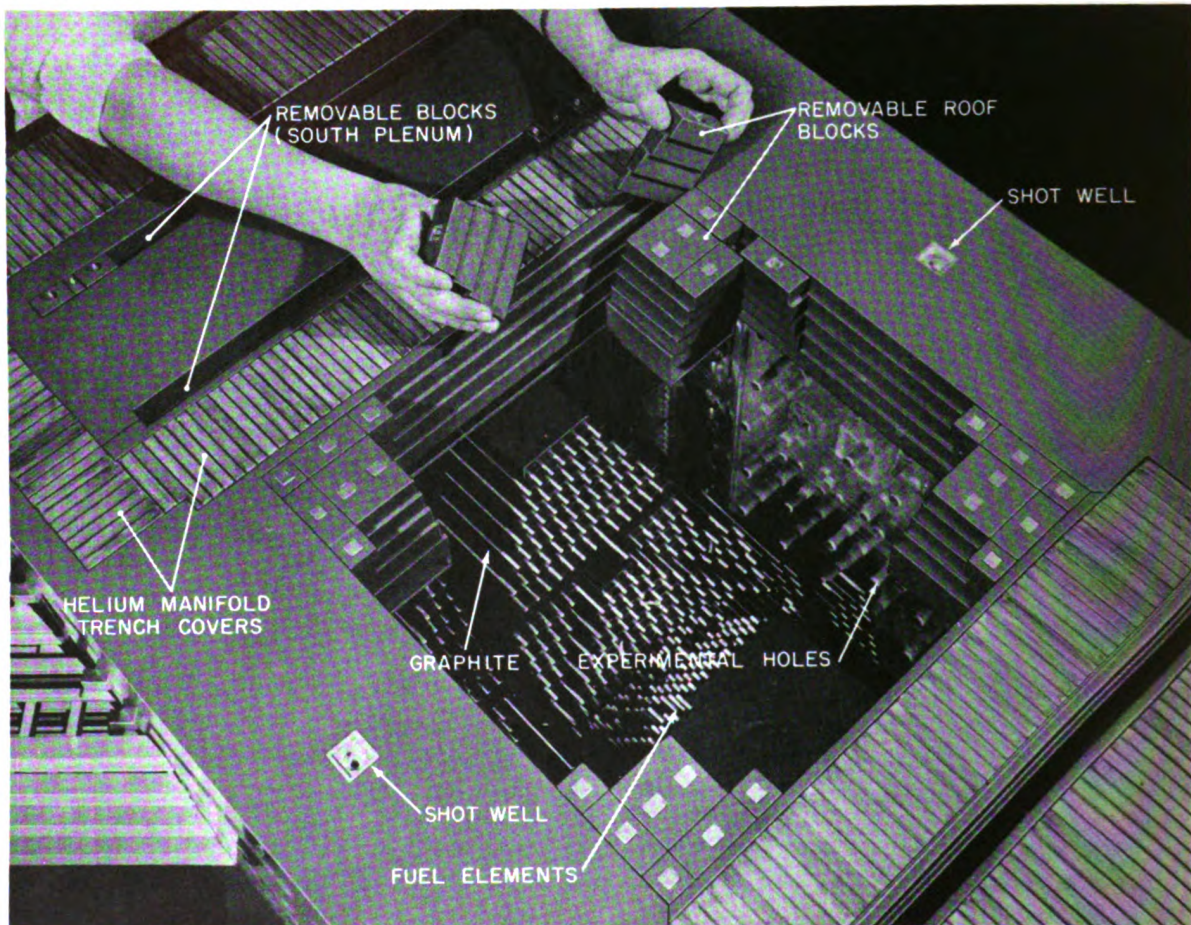


Fig. 6-4 Top view of reactor model. The black areas in the center represent the graphite; the white rods represent fuel elements. The stepped shielding blocks (on top) are removable for experimental use.

are left free for research. There are 16 horizontal rods, 2 driven by electric motors to regulate the power and 14 by hydraulic-electric systems to shut down the reactor. Any one of 56 signals in the emergency-shutdown (ESD) system will cause all the rods to be run in at full speed.

If electric power fails, the stored energy in constantly rotating flywheels on each motor shaft quickly drives the rods in. Additional safety systems are provided by (1) boron-steel shot which

teries to furnish power to instruments and lights when electricity is not available.

Experimental Facilities. The experimental facilities of the Brookhaven reactor include 30 horizontal 4- by 4-in. holes, each 25 ft long, which extend through the graphite at right angles to the fuel channels and through corresponding holes at the side shields. Graphite can be removed to make a 1-ft-square opening into the heart of the core.

Special facilities include pneumatically operated tubes, called rabbits, for whisking samples irradiated only a short time to nearby hot laboratories before radioactivity diminishes. For long-period irradiations there is an endless chain of sample holders extending through the reactor and the shield.

Two tunnels, one for instruments and one for animals, pass under the reactor to permit exposure near the center. When large areas of irradiation surface are desired, portions of the 20-ft-square section of the shield on top of the reactor can be removed in units of 4-ft-square blocks. In addition to manufacturing radioisotopes in quantity the reactor has produced several cobalt sources of radiation (500 curies to 5 kilocuries) for use by various laboratories.

Costs. Following is a breakdown of costs for the Brookhaven National Laboratory reactor.

Reactor.....	\$ 9,300,000
Reactor building.....	3,000,000
Air-cooling system, including fan hood, blowers, filters, air ducts, and air-cooling equipment.....	3,800,000
Reactor laboratories and hot chemistry laboratory.....	5,800,000
Site development, water-cooling tower, waste-disposal facilities, boiler house, electrical and service-distribution facilities, electrical substation and laboratory equipment.....	3,600,000
Total.....	\$25,500,000

Technical Data. Some pertinent technical data are summarized in Table 6-1.

Table 6-1 BNL Reactor Technical Data

Reactor size, graphite.....	25 ft ³
Lattice size, square array....	8 in.
Uranium diameter.....	1.1 in.
Area of channel.....	5.6 in. ²
Al/U ratio (by volume).....	0.2
Gap.....	2.75 in.
Metal surface temperature (max measured on pin center).....	360°C
Exit air temperature.....	170°C
Power output.....	30,000 kw
Thermal-neutron flux (max)...	5×10^{12} neutrons/cm ² /sec
Air flow.....	1,439,000 lb/hr
Pressure drop in reactor.....	63.4 in. H ₂ O
Pumping power.....	6350 kw

CORE DESIGN

Graphite. The graphite in the reactor serves two purposes: It acts as a moderator, which slows down the fast fission neutrons to thermal velocities, and it forms the actual structure of the reactor, which is penetrated by the channels in which the uranium metal is supported and through which cooling air flows.

Material and geometry. The graphite structure has the over-all dimensions of a 25-ft cube. It is divided into two halves by a vertical gap $2\frac{3}{4}$ in. (7.0 cm) wide and is built of 75 layers of 4- by 4-in. blocks of various lengths (up to 45 in. laid horizontally and parallel). All regular blocks are keyed to the blocks above and below by graphite keys. These keys run horizontally from east and west and prevent misalignment due to various forces acting on the structure. A total of 58,911 blocks, weighing 1,463,550 lb, were machined on standard milling machines equipped with carbide-tipped cutters. An isometric view of the graphite structure is shown in Fig. 6-5. The material was made available by the AEC and was classified into four grades (A, B, C, and D) based on diffusion length (see Table 6-2). Figure 6-6 shows the locations of the various grades of graphite; several graphite shapes are illustrated in Fig. 6-7.

Gap. The use of a gap, or air-feed slot, is for the purpose of reducing the power required for pumping cooling air. This is accomplished by essentially halving the length of path of air flow within the channels and doubling the cross section of air-flow path. The width of the gap is a compromise between theoretical physics and air-flow considerations; the neutron escape, and hence the reactor size, increases with increasing gap width, while the pressure drop across the gap, and hence the pumping power, decreases with increasing width. The geometry of the gap is designed to minimize the escape of neutrons for a given gap width.

Each of the two faces forming the gap is cut into eight triangular sections. Each section contains 171 channel openings. Alternate triangular sections are offset $2\frac{3}{4}$ in. (7 cm) so that each gap face appears as a surface of eight triangles, alternately raised in relief or recessed. If the two halves were moved together, the faces would interlock, with a 1-in. lateral clearance left between

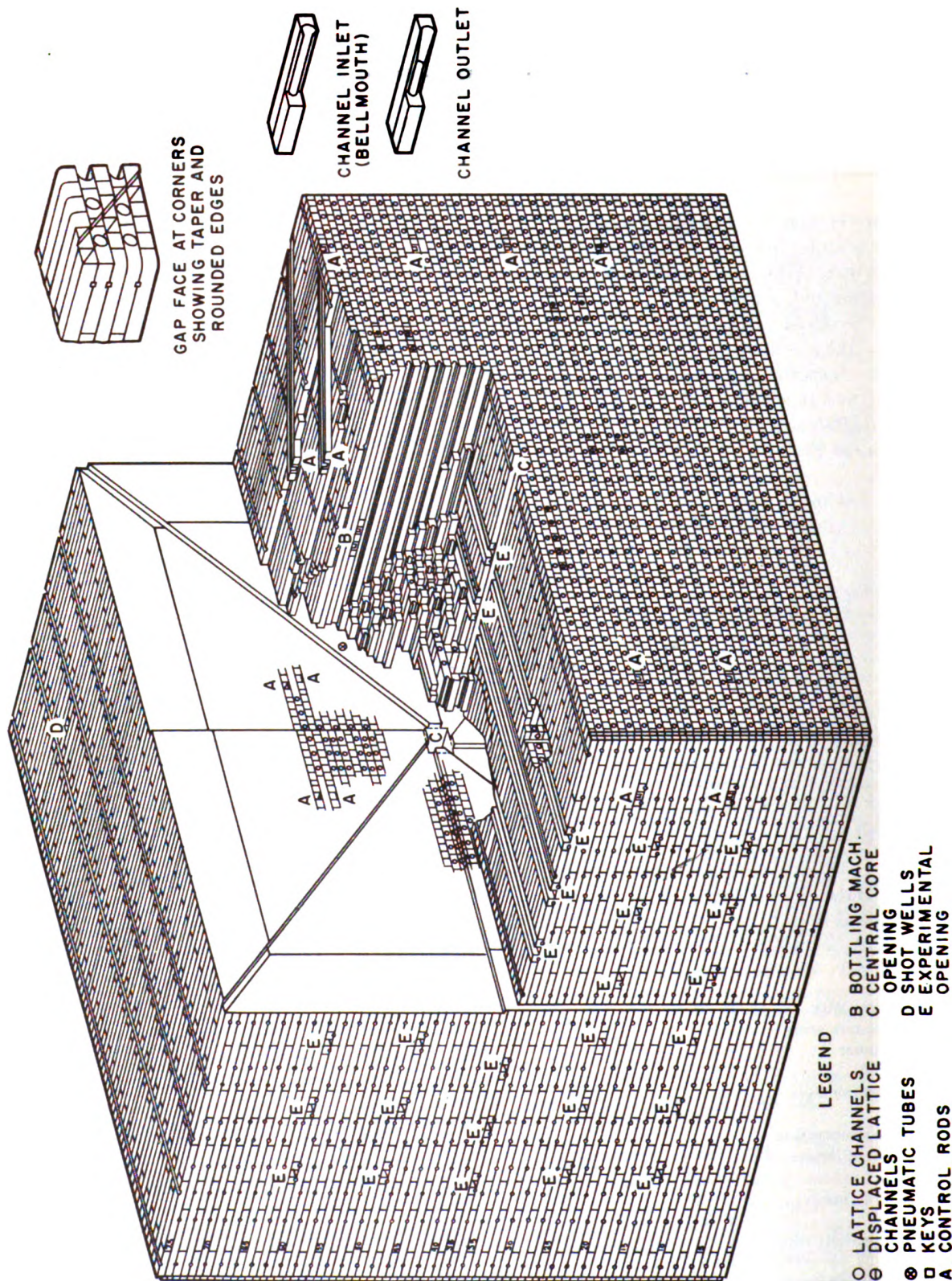


Fig. 6-5 Isometric view of graphite structure.

Table 6-2 Classification of Graphite

Type	Zone in reactor	Chemical analysis		Density, g/cm ³	Diffusion length, cm			Weight in each zone, lb
					Small sigma reactor		Adjusted to basis of large sigma reactor	
		Boron, ppm	Ash, ppm		As measured	Corrected to 1.60 g/cm ³		
AA	..	0.1	10	1.656	53.67	55.54	13,060
CS	A	1.641	52.07	53.4		
AGOT	A	0.5	350	1.686	51.12	53.86	52.96	179,820
AGOT	B			1.703	49.74	52.94	52.05	170,390
AGOT	C			1.702	48.56	51.64	50.77	147,105
AGHT	D	0.7-0.8	400-450	1.636	48.03	49.09	953,175
Total	1,463,550

the "tongues" and the "grooves." The gap is $2\frac{3}{4}$ in. wide, but starting at a distance of $2\frac{1}{2}$ ft from the gap entrance, it flares to a width of $4\frac{1}{3}$ in. The edges are rounded to a $3\frac{1}{2}$ -in. radius.

Channels. The channels which carry the fuel assemblies and the cooling air run north and south through the reactor. There are 36 rows of 37 holes each and 1 row of 36 holes; the center hole

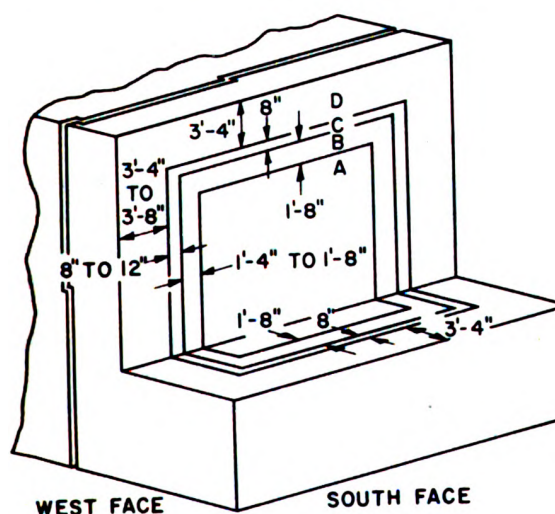


Fig. 6-6 Locations of grades of graphite.

The gap is spanned by a 12-in.-square removable graphite core, which fills a 12-in.-square opening running along the axis of the reactor. There are also two emergency shot wells in the gap, consisting of aluminum tubes which enter at the two upper corners and run down along the diagonals to the core.

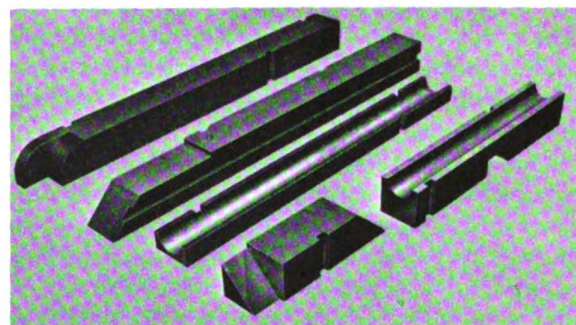


Fig. 6-7 Typical graphite shapes.

in the eighteenth row is omitted, since this space is occupied by the 12-in.-square removable core. The total number of channels is therefore 1368. They are symmetrically arranged on 8-in. centers in a square array starting 6 in. from the reactor edges. The channels have a circular cross section 2.67 in. in diameter and were made by a milling cutter with a 1.335-in. radius in two adjacent blocks. The circular edges between blocks are chamfered (0.005 in. and 45°) to avoid the possibility of a shoulder, caused by shifting of the graphite, that might cause blocking of the aluminum cartridges as they are pushed through

the channel. The channel ends are expanded in two ways: (1) The inlet mouth (at the gap) has a 3.25-in. radius tangent to the cylindrical surface at a point $1\frac{25}{32}$ in. from the end; and (2) the outlet has an over-all 7° taper starting $6\frac{7}{8}$ in. from the plenum end of the channel, thus having a terminal diameter of $3\frac{1}{2}$ in. (see Fig. 6-5).

Each channel in each half of the reactor contains a ring-shaped groove to receive the anchoring projection of the cartridge. According to calculation an unanchored cartridge (weight about 90 lb) would start moving at a pressure differential of 80 in. H_2O .

Several of the outer rows of channels are not charged with metal, and these channels are filled with graphite plugs.

Displaced channels. In addition to the symmetrical square lattice of 1368 channels, there are 30 channels identical in dimensions with the other channels but displaced from the lattice positions by a distance of 4 in. Each of these channels is a companion to one of the regular channels in the lattice, and one channel of each such pair is loaded with metal, the other being plugged with graphite in the same manner as the unused outer channels. The purpose of the displaced channels is to provide regions of localized high flux density of slow neutrons by increasing the distance between adjacent cartridges.

Channel-outlet restrictions. The channel outlets are restricted by varying degrees in order to proportion the amount of air flowing through a given channel to the amount of heat liberated in that channel. The channels are divided into sixteen groups or stages, which are concentric about the reactor core. The channels in the central stage have no outlet restrictions, while the size of restrictions of the other stages increases from stage to stage when moving toward the outside of the reactor. The channels in a given stage, however, have identical restrictions.

The design of the channel restrictions was guided by (1) ease of installing and removing, (2) reproducibility of behavior, (3) minimum effect of rotating restriction, and (4) ease of duplication.

Graphite growth. As a result of neutron bombardment, the crystalline structure of the graphite is distorted. This distortion is cumulative and manifests itself as "graphite growth" in a

direction perpendicular to the direction of extrusion of the graphite. (A negligible fraction of this growth is in a direction parallel to the direction of extrusion.) The direction of extrusion of the blocks in the reactor runs along the north-south axis of the reactor. Consequently, the problem of graphite growth is confined to the east, west, and top faces of the reactor. Maximum growth occurs at the center of the reactor in the region of highest neutron flux and decreases toward the outside of the reactor.

The main problem presented by graphite growth is the fact that in the upper half of the reactor—above the thirty-eighth layer—each successive layer grows less than the one just beneath. However, because of friction between layers, the upper layers tend to be carried outward by the lower ones, opening gaps between the blocks which permit cross leakage of air between the channels.

The provisions made in the reactor against the effects of graphite growth are aimed primarily at preventing the graphite blocks in the upper layers from being carried outward (by friction) in the east and west directions. There is no provision for restraining the growth itself.

The top of the reactor supports the removable blocks of concrete shielding, which constitute a considerable and constant load tending to prevent gaps between the graphite blocks due to vertical growth. The east and west faces of the reactor have been spring-loaded with forces sufficiently large to overcome the calculated frictional forces that may tend to move the blocks outward. These restraining forces will permit the lower (faster growing) of two layers in the upper part of the reactor to slip under the upper layer without carrying it along. In calculating the frictional forces on a given layer, it was assumed that the force on the lower surface tending to carry it outward is a function of the static coefficient of friction, while the restraining force on the upper surface is a function of the sliding coefficient of friction, as determined for normal rates of motion. At the infinitesimal rates of motion involved in graphite growth, it seems likely that the two coefficients should have nearly the same value, but since no experimental information was available for conditions of very slow movement, the above procedure, which gives a very conservative answer, was used. On this basis the springs are de-

signed for a preloading of 450 psf on the east and west faces of the reactor above the center line. A much lower loading force is used below the center line. Figure 6-8 shows one such spring in the



Fig. 6-8 Loading spring in the vicinity of the reactor gap.

vicinity of the reactor gap. The keying along the bottom of the reactor is also shown.

Foundation. The construction of the reactor foundation was complicated by the design requirement that the control rods should remain operable

in the event of a severe earthquake. The AEC Reactor Safeguard Committee recommended in addition that other control devices, boron shot and trichlorobenzene, be added to shut down the reactor if the control-rod system failed. The foundations and building were designed to resist a horizontal shock equivalent to 0.1 gravity. Table 6-3 lists the foundation loads. Figure 6-9 is a

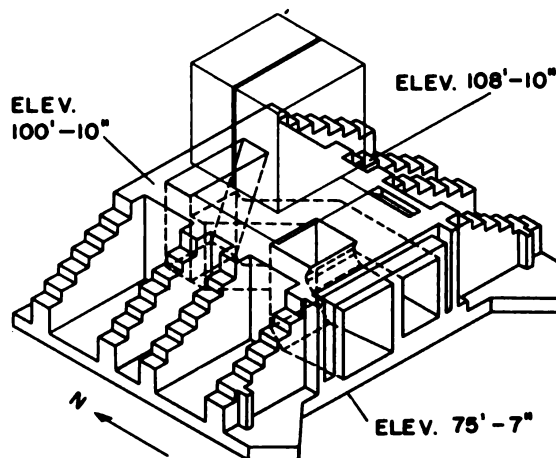


Fig. 6-9 Sketch of reactor foundation.

sketch of the reactor foundation, and Fig. 6-10 is a view of the foundation during construction.

Table 6-3 Foundation Loads

Foundation	Load, lb
Reactor:	
Concrete slab	4,064,000
Concrete between slab and elevation 110 ft.	12,846,000
Backfill	6,080,000
Reactor shield	9,820,000
Graphite and uranium	1,700,000
Miscellaneous	1,866,000
Total dead load	36,376,000
Maximum floor load	4,030,000
Total live load	40,406,000
Duct:	
Concrete slab	939,000
Concrete duct	1,065,000
Concrete for building column footings and elevator pit	1,414,000
Building column load	1,354,000
Backfill	962,000
Miscellaneous	453,000
Total	6,187,000

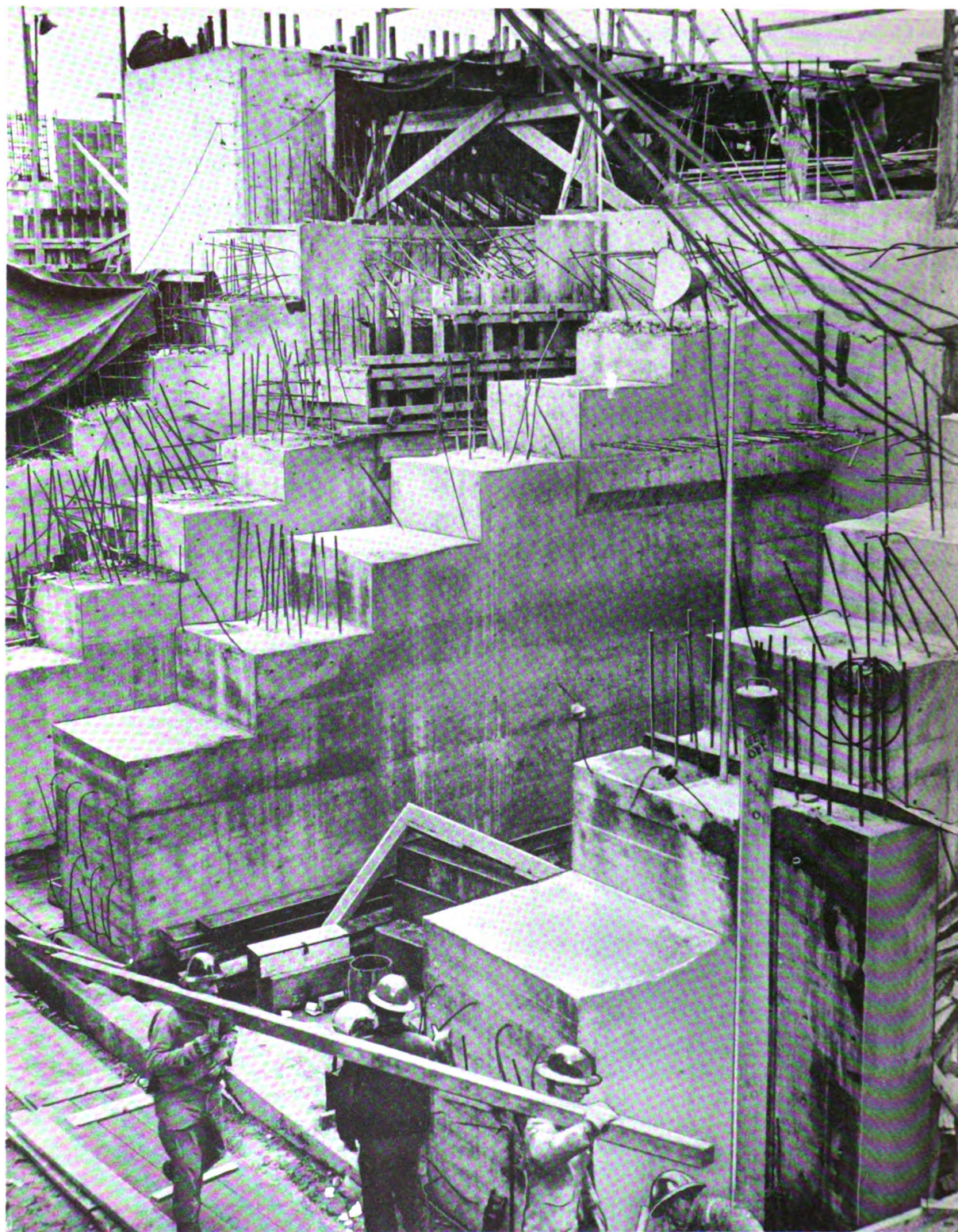


Fig. 6-10 Reactor foundation during construction.

Fuel-element Design. The fuel elements for the Brookhaven reactor are internally anodized extruded and drawn aluminum cartridges containing 1.1-in.-diameter cylindrical slugs of natural uranium. The slugs are loaded into a single aluminum cartridge running half the length of the

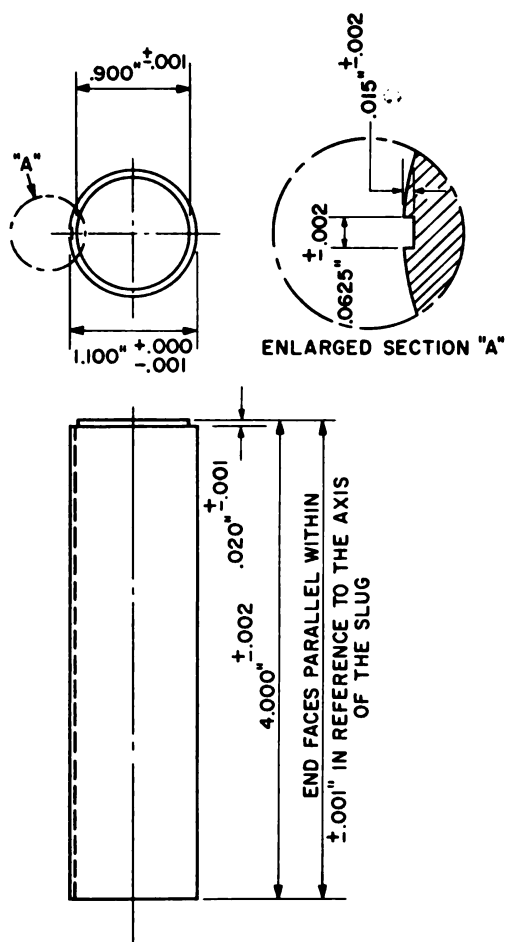


Fig. 6-11 Dimensions of uranium slug.

reactor in order to reduce welds and also to simplify the helium leak-detection system. The cartridges are finned to increase heat transfer, thus making it possible to locate them close to the center of the channel. Because of this centering, the heat-transfer area can be used more efficiently, the range of aluminum temperature at a given cross section is narrower, a higher neutron flux can be tolerated, and the cooling air can be heated more. Six equally spaced longitudinal fins are employed.

Uranium slug. Figure 6-11 shows a sketch of a uranium slug with its dimensions and tolerances. Thirty-three slugs are loaded in a single cartridge 11 ft long and enclosed in a helium atmosphere. To provide continuous passage for helium through the cartridge, one end of each slug has a button of 0.900 ± 0.001 in. diameter centered on the end face and raised 0.020 ± 0.001 in. The slug also has a slot 0.0625 ± 0.002 in. wide and 0.015 ± 0.002 in. deep running the full length. The dimensions of the slot and button were chosen to give a definite desired volume of helium gas in the cartridge. When the reactor attains operating temperature, the differential expansion of aluminum with respect to uranium will increase the volume occupied by helium so as to maintain a substantially constant helium pressure. Hence, there is no tendency of the helium gas to surge into or out of the cartridge.

Amount of uranium. The requirements for 33×33 loading and 35×35 loading are given in Table 6-4.

Table 6-4 Required Uranium Quantity

Item	33 × 33 loading	35 × 35 loading
Uranium slugs per aluminum cartridge.....	33	33
Number of cartridges.....	2176	2448
Total number of slugs required.....	71,808	80,784
Weight per uranium slug....	2.58 lb	2.58 lb
Total weight of uranium required.....	185,000 lb (92.5 tons)	208,000 lb (104.0 tons)

Cartridge. Figure 6-12 shows the finned aluminum cartridge section which houses the uranium slugs. Type 2S aluminum is employed and the inside of the cartridge is anodized because of the reaction of uranium and aluminum at the interface temperature of 350°C . The requirements for the aluminum-cartridge geometry arise out of nuclear physics studies and practical considerations of cooling and fabricating methods. The design provides for the highest rate of heat transfer with the least increase in pressure drop and in aluminum-to-uranium volume ratio.

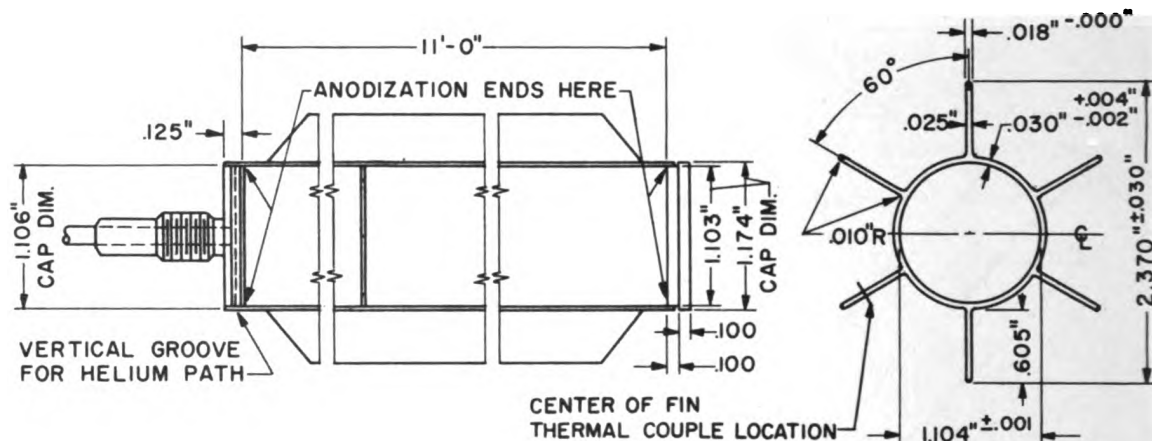


Fig. 6-12 Finned aluminum cartridge and cross section.

The aluminum of the cartridge may contain up to about 1 per cent total impurities, as indicated in Table 6-5.

Table 6-5 Aluminum Cartridge Impurities (Maximum)

Element	Amount, %
Copper.....	0.20
Silicon plus iron.....	1.00
Manganese.....	0.05
Zinc.....	0.10
Other elements (each).....	0.05
Other elements (total).....	0.15

Figure 6-13 shows a section of the aluminum cartridge as extruded and after drawing.

Canning of uranium. The procedure for canning the 33 uranium slugs in the aluminum cartridge was developed at the Brookhaven National Laboratory, and the processing was carried out at the site by Brookhaven personnel. The sequence of operations involved in canning is as follows:

1. Facing and trimming the end of the cartridge.
2. Chemical stripping of anode coating from the trimmed end of the cartridge.
3. Argon arc welding of the cap into the stripped end of the cartridge.
4. Degassing of the welded tube in a vacuum at 350°C.
5. Helium mass-spectrometer leak testing of the first closure and entire tube.
6. Trimming to length and chemical stripping of oxide film on either end of the cartridge.

7. Insertion of the cartridge into the storage tube and the loading of slugs.

8. Final trimming of the cartridge.

9. Argon arc welding of the closure cap (which previously had been attached to the helium tube by brazing) and leak testing.

10. Leak testing of the assembled cartridge.

11. Compressing the cartridge onto the slugs.

12. Final leak testing of the assembly.

13. Flame brazing of anchor to fins.

Helium System. The helium system enables the rapid detection of cartridge leaks and contributes to the effectiveness of the cooling system, since its thermal conductivity is five times that of air. Each uranium slug is surrounded with helium under pressure to prevent oxidation of the metal in case of a cartridge leak. Furthermore, each cartridge is connected by tubing into a manifold system which, by means of a pressure indicator, permits automatic determination of the vertical row in which the defective cartridge is located. By rearranging valves, the horizontal row may also be identified, thus locating the faulty cartridge. Figure 6-14 is a schematic diagram of the system, where for simplicity only 3 of the 37 cartridges in each vertical row and only 3 of the 37 rows in each half of the reactor are shown. The leak-detection system is theoretically responsive to a leak of $\frac{1}{2}$ cm³ of helium. Normal changing pressure is 1 psig.

Breathing. The requirement that "breathing" (surging) of the helium system be kept to a mini-

mum effectively fixed the volume of the gas space in the cartridge. It can be shown that for a sys-

where t_2 = operating temperature, °C

t_1 = canning temperature, °C

V_g = gas volume built into assembly (i.e., slot + annulus around button)

V_a = difference in volume coefficients of aluminum and uranium

This reduces to

$$V_g = V_a(3\alpha')(273 + t_1)$$

It is seen that V_g depends only on the canning temperatures and the difference in volume coefficients and not on the power level.

A volume of 16.4 cm³ would give zero breathing. The design is based on slight outward breathing so that during reactor cooling, helium flow is toward the cartridge, thus preventing the spread of fission products. A volume of 18 cm³ gives a pressure build-up not exceeding 3 psig upon heating a sealed cartridge from 25 to 350°C. Slot and button dimensions were selected to give a volume of 5.4 cm³ for 33 slugs, which, with the 13 cm³ maximum due to spring-back, gives the desired total. The volume of helium in the cartridges in each half of the reactor for a 37 × 37 lattice loading is approximately 0.9 ft³. For the entire helium system the helium volume is about 5.2 ft³.

Purging and charging systems. From time to time the helium system must be purged to remove impurities that might contaminate the system or cut down the rate of heat transfer in the cartridges. Helium and gaseous fission products are evacuated through a carbon cold trap which contains activated coconut char and is surrounded by liquid nitrogen refrigerant in a Dewar flask. The system is purged during periods of reactor shutdown. Figure 6-15 shows the arrangement of the conduit carrying helium tubing on the north face of the reactor, while Fig. 6-16 illustrates the manifolding and the location of instruments in the trench on top of the reactor.

Construction materials. The valves are of the 1/4-in., packless type having a bellows seal, and they are made of nickel, monel, and copper. The conduit is made of stainless steel. The tubing inside the reactor consists of aluminum while the tubing on the outside of the reactor is made of copper.

Permanent copper-to-copper joints are sweat

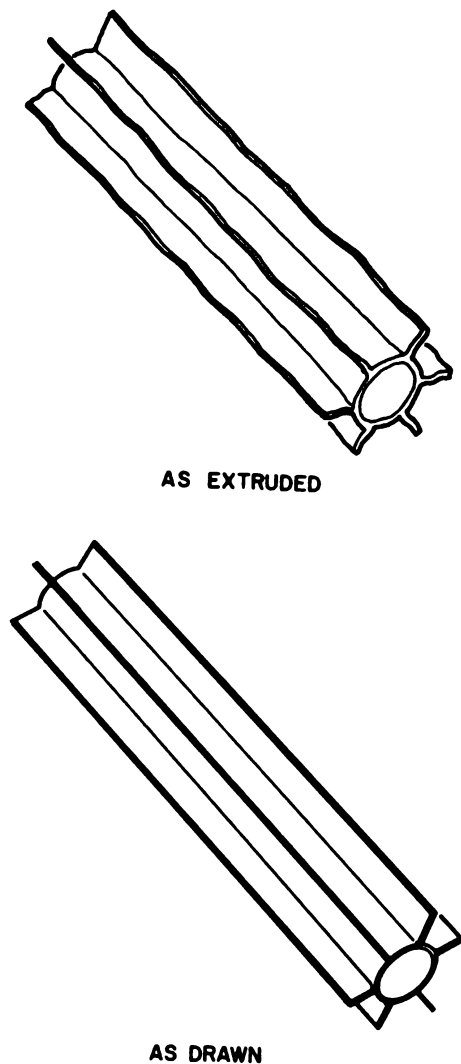
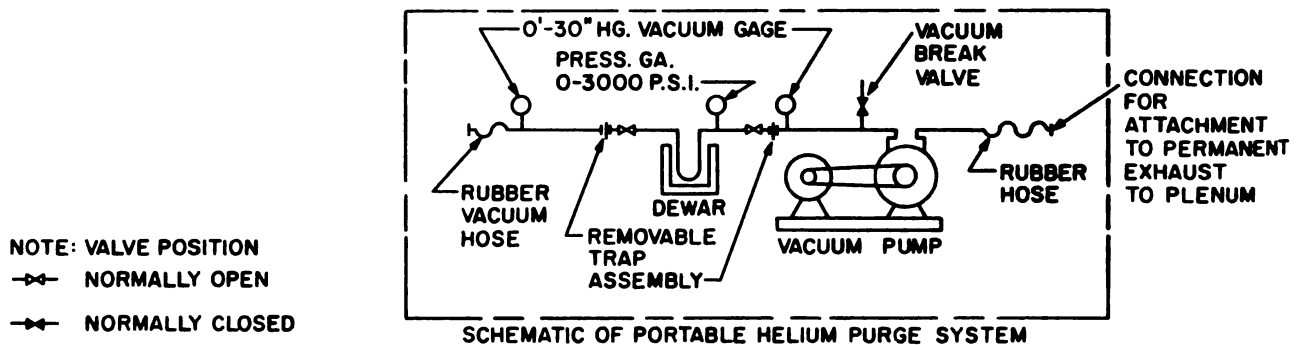
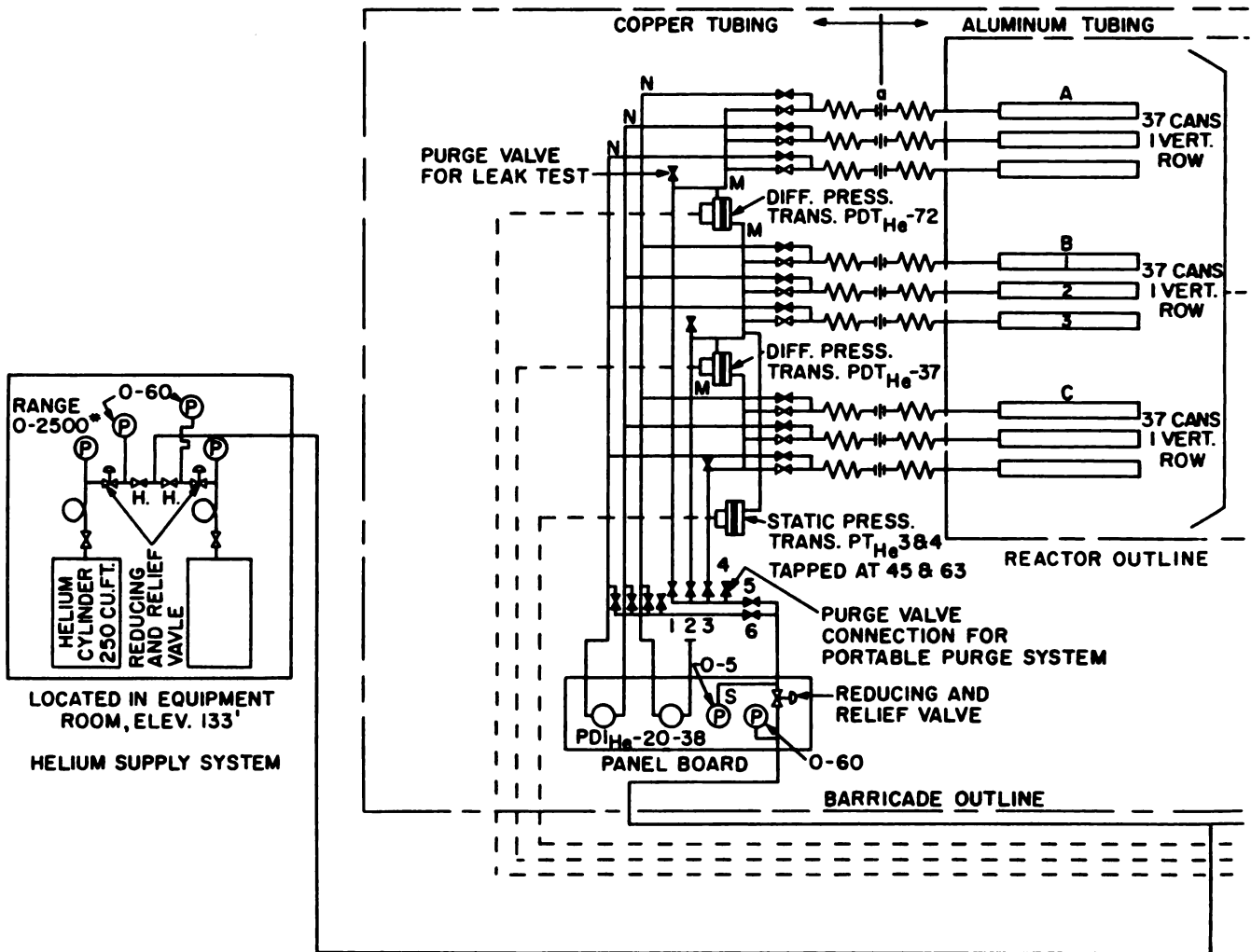


Fig. 6-13 Aluminum cartridge, as extruded and drawn.

tem at uniform temperature and constant external pressure¹ there will be no gas surging if

$$V_g \times \frac{(273 + t_2)}{(273 + t_1)} = V_g + V_a(3\alpha')(t_2 - t_1)$$

¹ Actually the temperature of the helium in the aluminum capillaries increases during reactor operation, tending to increase the external pressure slightly.



SCHEMATIC OF PORTABLE HELIUM PURGE SYSTEM

Fig. 6-14 Schematic

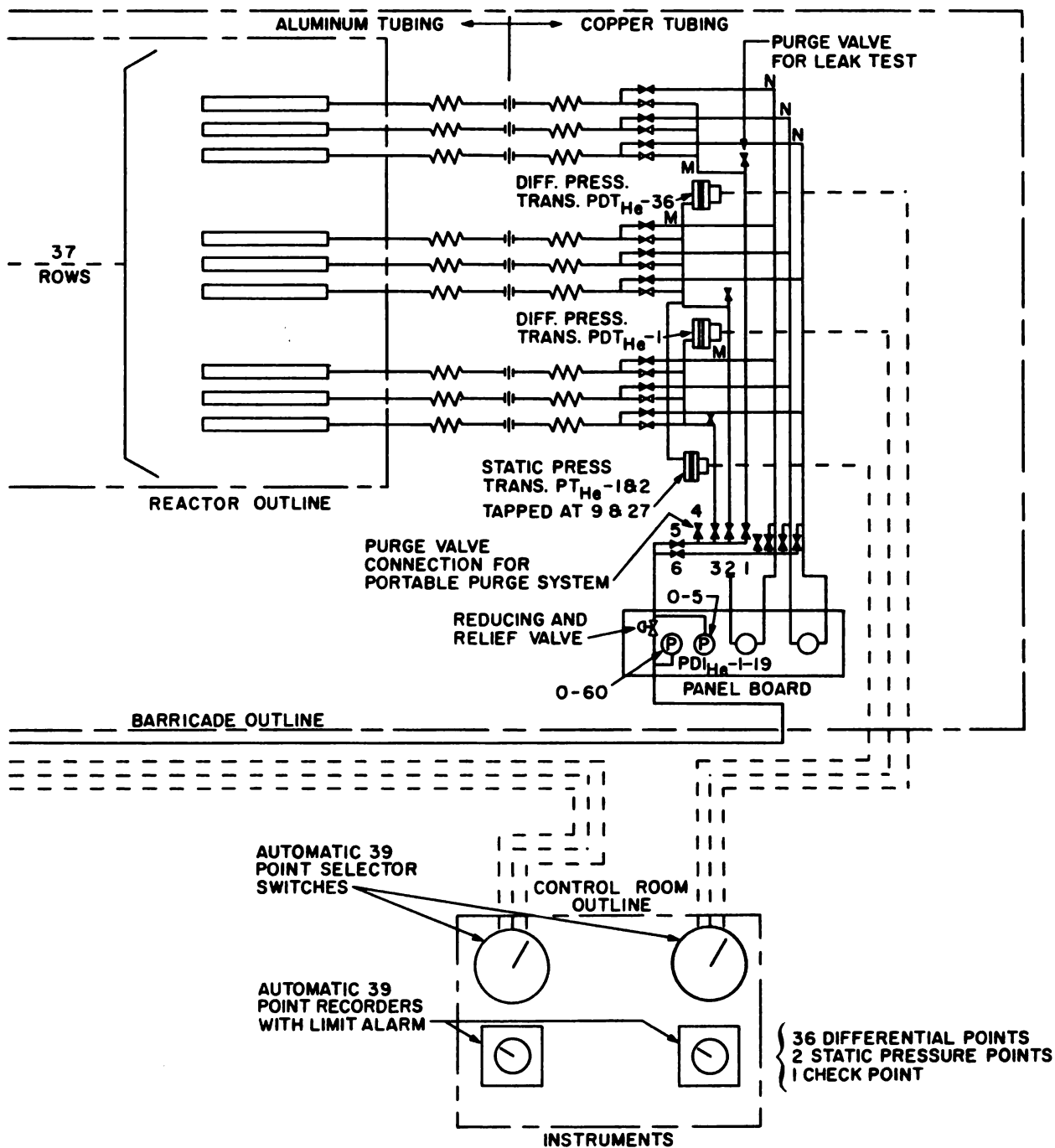


diagram of helium system.

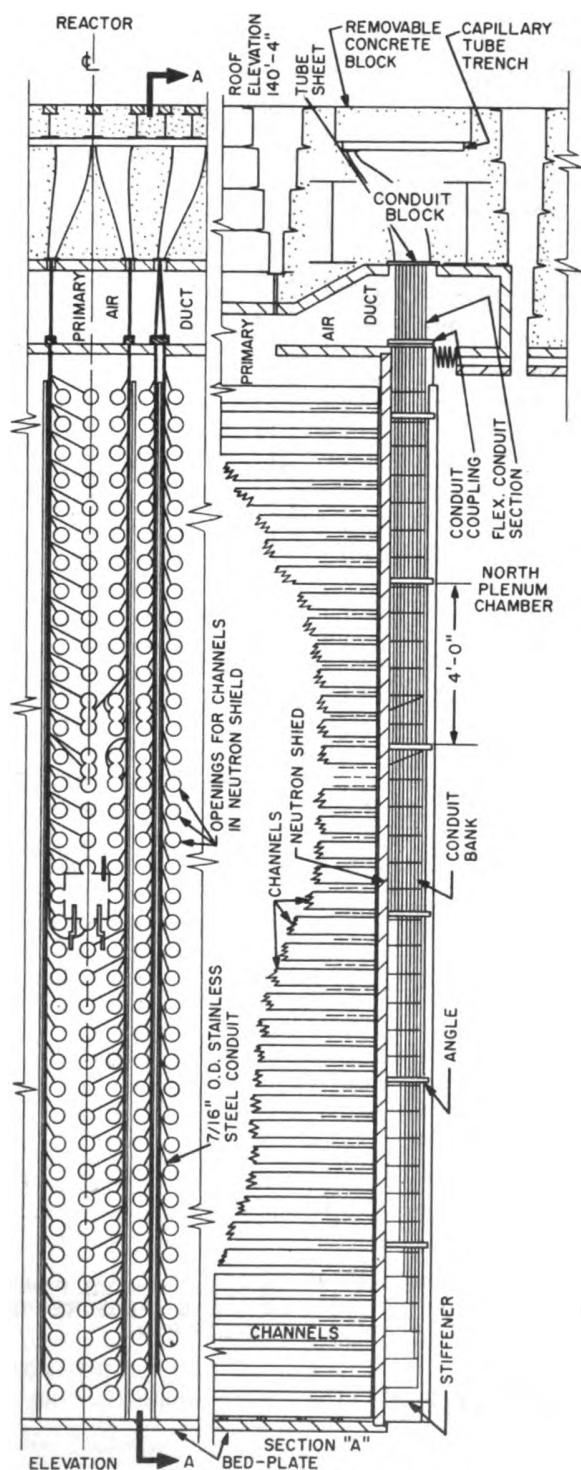


Fig. 6-15 Conduit arrangement of helium system.

fittings sealed with phospho-copper. Permanent brass-to-copper joints are sweat fittings sealed with silver braze. Removable joints consist of compression-flare fittings of the Parker triple type.

The entire system was leak-tested with a mass spectrometer to be less than 20 micron ft^3/hr . Each joint or component was tested to be within 5 micron ft^3/hr .

Instrumentation. The helium leak-detection system is provided with instruments which automatically scan and, by electrical transmission, remotely indicate and record the differential pressures between vertical rows of cartridges. An alarm system is actuated by an extreme differential. Subsequent observation of the record discloses the vertical row containing the faulty cartridge. In addition a manifold system allows measurement of differential pressures between horizontal rows of cartridges. The differential pressures are locally indicated by panel-mounted gauges; this permits positive identification of the leaky cartridge.

The sensitivity of the apparatus to differential pressure is limited to expansions of 25 in. H_2O , maximum, and may be adjusted to 2.5 in. H_2O , minimum. The change in sensitivity changes the range of the differential-pressure units but does not affect the range of the static-pressure units. A change of differential pressure of 0.06 in. H_2O will cause pen motion.

SHIELDING

The reactor shield was designed on the basis that personnel should not receive more than 5 mr during 8 hr of continuous exposure. This is one-tenth of the tolerance value adopted for the Brookhaven Laboratory, which in turn is one-half the tolerance value that has been used at a number of sites. The shield adequacy has been proved.

Since the shield design was based on a concrete density of 225 lb/ft^3 and the average density obtained was 279 lb/ft^3 , it was necessary that an additional margin of safety against radiation be provided.

Specific requirements for the high-density concrete were a minimum density of 225 lb/ft^3 , a limonite-scrap-iron ratio of 1:1 by absolute vol-

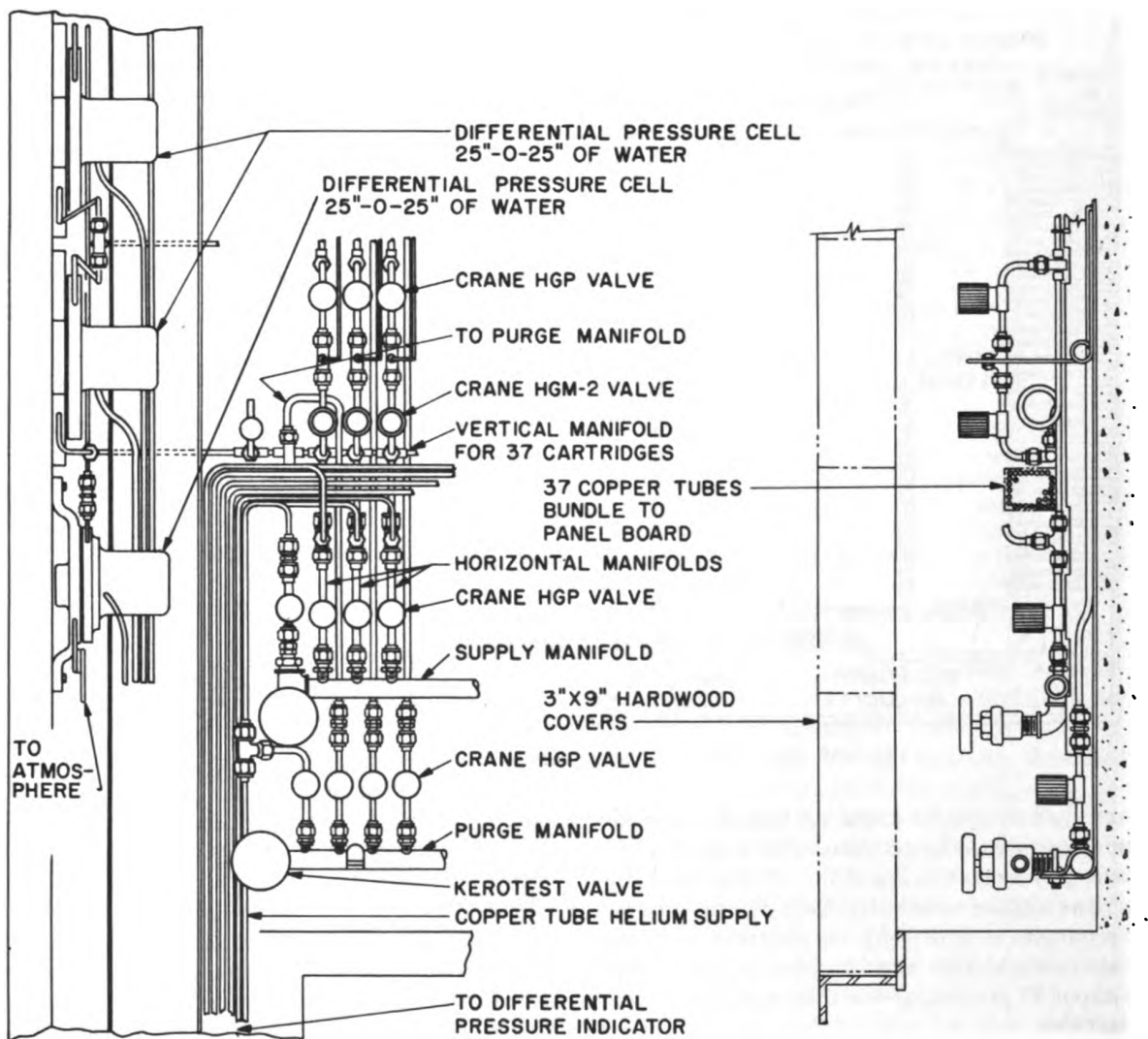


Fig. 6-16 Manifolding and instruments in trench.

ume, and satisfactory structural properties. (Limonite is an iron ore with a relatively high water of hydration.) It is necessary to maintain the temperature reached by the concrete during reactor operation below 212°F in order to prevent the dehydration of the water contained by the limonite.

General Description. Figure 6-17 shows a simplified plan view of the reactor, plenum chambers, and the shield. The reactor shielding is 55 ft long, 37 ft 6 in. wide, and 33 ft 7 in. high. It ex-

tends from an elevation of 106 ft 9 in. (3 ft 3 in. below the main-floor elevation) to an elevation of 140 ft 4 in. Proceeding from the graphite outward, the shield consists of 6 in. of steel plate for thermal-neutron absorption, 4¼ ft of high-density concrete, and an outer casing of 3 in. of steel plate for structural purposes. The 6-in.-steel neutron shield is supplemented with a band 1 ft thick and 20 in. wide girdling the reactor around the gap, where the flux is higher. The east, west, and top faces of the graphite reactor are covered with close-fitting membranes of ¼-in. steel or ½-in. alumi-

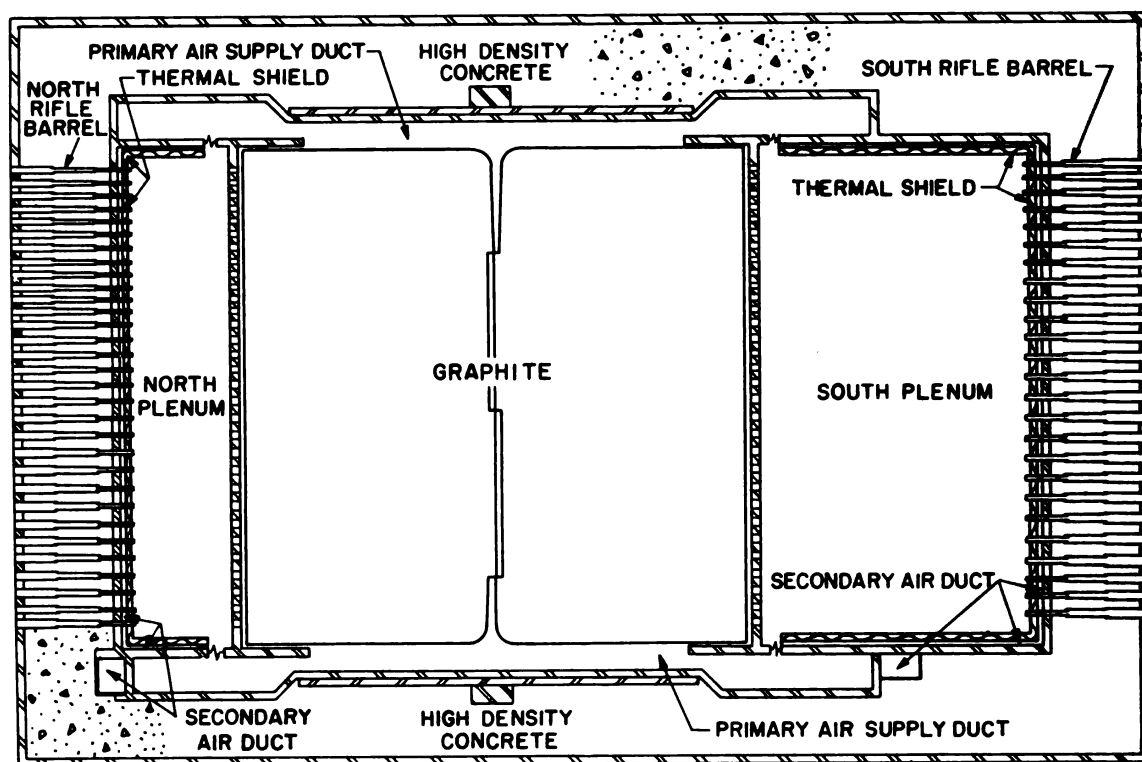


Fig. 6-17 Plan view of reactor showing plenum chambers and shield.

num plate to provide a seal between the inlet air and the seams in the graphite. The airtight membrane may be seen in Fig. 6-18. These plates also serve as neutron shields, but their effectiveness for this purpose is limited by the fact that each one contains about 1400 circular openings, amounting to about 25 per cent of the total area, for the air channels.

The shield is penetrated by a number of openings, required either for loading or unloading the fuel cartridges or for experimental purposes. Figure 6-19 is an isometric cutaway view of the reactor shielding with some features omitted for clarity. Figure 6-20 shows a section through the shield.

Loading and unloading is done through the "rifle-barrel" openings in the south-plenum wall. Corresponding openings, but of smaller dimensions, are provided in the north-plenum wall for emergency pushing of fuel cartridges and other auxiliary use. Most of the experimental openings are in the east and west faces of the shield, matching openings running through the graphite. The

inlet air ducts for these openings are spanned by stainless-steel bellows between the shield and the $\frac{1}{4}$ -in. plate next to the graphite; these flexible tubes seal the inlet air from the experimental openings. The rifle barrels and the experimental openings in the shield have a two-step design to prevent straight-through radiation leakage and are filled with suitably designed plugs of steel, concrete, and graphite.

On the top face of the reactor, a 20-ft square in the shield is made of removable plugs which can be taken out for experimental purposes. Each plug rests directly on the graphite and its bottom section is made up of steel I beams and plates so as to form part of the air inlet duct when the plugs are in place. Other openings in the shield are the control-rod openings, the east-west "scanning holes" in the roofs of both plenums, openings for radiation instruments used in control and monitoring, and openings for pneumatic tubes in the north face of the reactor. These openings and several others will be described later.

The dense concrete used in the shield has to be

kept below 212°F to prevent dehydration, which would cause both physical deterioration and loss of shielding power against fast neutrons. On the four lateral faces of the reactor the concrete is protected by the cooling action of the primary air. A secondary air-cooling system is provided to protect the shielding surrounding the two plenum

exceed 200°F. The position of this thermal shield is shown in Fig. 6-19.

High-density concrete. The high-density concrete was made with limonite as the fine aggregate and scrap iron with a particle size range of $\frac{1}{2}$ to $1\frac{1}{8}$ in. as the coarse aggregate, plus portland cement and a minimum of water. The upper limit for the iron-particle size was prescribed by the close spacing between rifle barrels and other tubes passing through the shield.

The high-density concrete used in the shield contained the ingredients shown in Table 6-6.

Table 6-6

<i>Ingredient</i>	<i>Amount per cubic yard</i>
Portland cement.....	940 lb
Limonite.....	1880 lb
Steel or iron punchings.....	4100 lb
Plastiment.....	15 lb
Water.....	35.5 gal

The limonite was specified to contain a minimum of 50 per cent iron and a minimum of 8 per cent chemically combined water (retained at 100°C). The high-density concrete described above was used throughout the shield proper, including the large removable roof plugs. This concrete could not be used in the removable plugs for the rifle barrels and the experimental openings because of their small dimensions. A high-density grout was developed for this purpose. The grout was prepared by substituting steel "grits" and granules for the coarse steel aggregate. The proportions of the ingredients were also modified slightly, as shown in Table 6-7.

Table 6-7

<i>Ingredient</i>	<i>Amount per cubic yard</i>
Cement.....	752 lb
Limonite.....	1690 lb
Steel grits.....	2280 lb
Steel granules.....	2100 lb
Plastiment.....	12 lb
Water.....	50 gal

Radiation tests during pouring. To ensure an absence of voids in the concrete in the narrow spaces between the rifle barrels in the north and

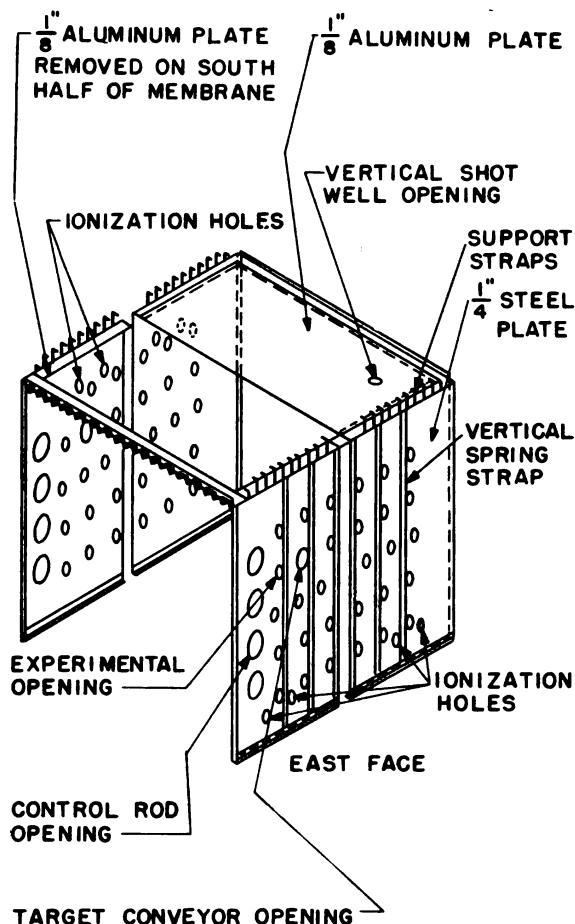


Fig. 6-18 Construction of airtight membrane.

chambers, which contain air at approximately 428°F. Essentially this consists of a $2\frac{1}{2}$ -in. air space lining the plenum chambers, lying between the plenum walls proper and an insulating partition known as the thermal shield. The thermal shield is $4\frac{1}{2}$ in. thick and is made of six or eight layers of crimped aluminum sheets between two steel binding plates. The secondary air stream flows through this $2\frac{1}{2}$ -in. space at a rate sufficiently high so that its exit temperature does not

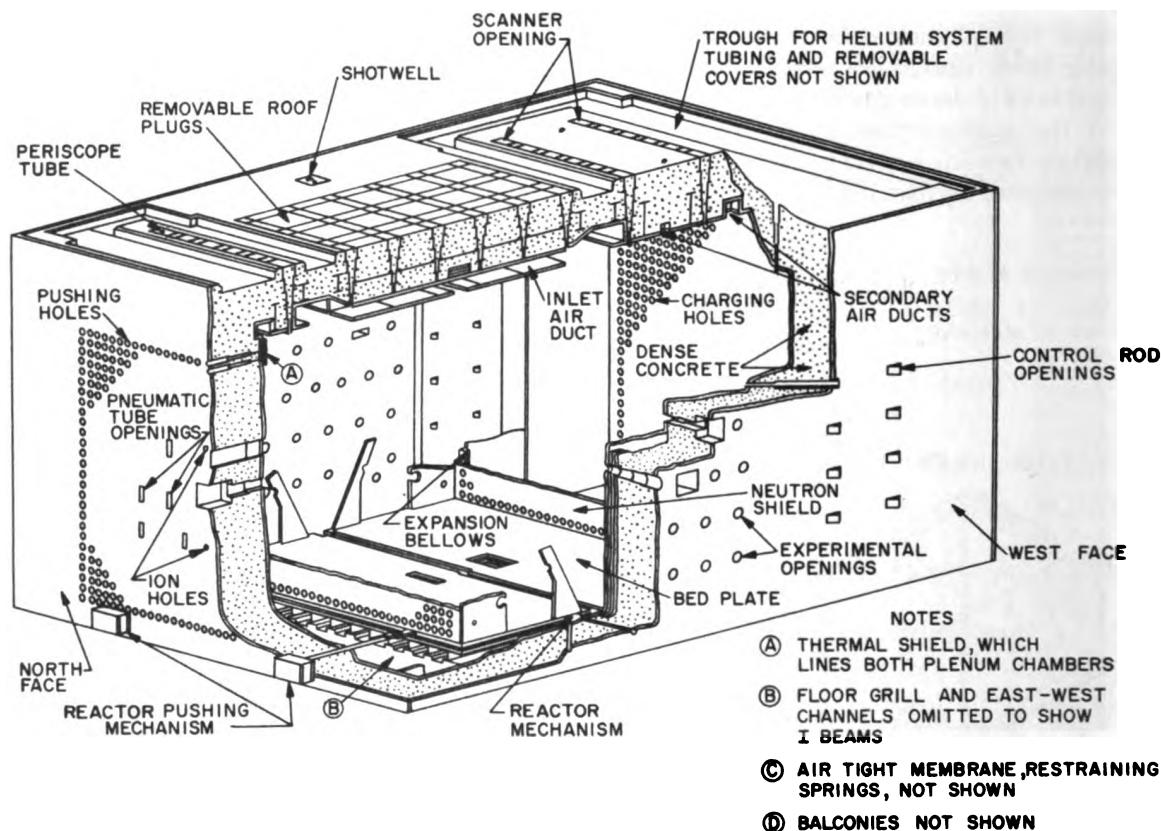


Fig. 6-19 Cutaway view of reactor shield.

south walls, a special inspection procedure was used during pouring. It consisted of measuring the intensity of gamma radiation reaching a rifle barrel from a 5-mg radium source inserted into a vertically adjacent rifle barrel. The source was kept as small as feasible to minimize radiation hazard to the inspectors and workers, and the resulting counting rate was about 500 counts/sec. At this low rate the "normal" reading fluctuated over a moderate range, but a void in concrete was clearly indicated by an increase in the meter reading of 15 to 20 per cent. When such an increase was observed, the concrete was vibrated until the reading fell back to normal throughout the length of the rifle barrel.

Steel casings. The shield was designed with inner and outer steel walls which held rifle barrels, experimental holes, and other sleeves in place and served as forms for the concrete. The 3-in. inner plate served to absorb slow neutrons and dissi-

pate absorption heat to the air streams. The use of 3-in. steel plate for the outer wall was partly influenced by availability of this material.

Fabrication. The shield was fabricated in sections. Each section, called a *crate*, was from 4 to 6 ft high and 8 to 14 ft long, with front and back steel plates held together by angle irons and with all the holes, tubes, and liners installed. Each side of the shield contained seven tiers of crates. On the east and west walls, the inner plates were displaced horizontally and vertically 1 in., which offset them from the outer plates, preventing possible radiation leakage through the joints. These plates were flame-cut from large plates. Holes for the experimental and other openings were also flame-cut. By maintaining extreme care a tolerance of $\pm 1/32$ in. was maintained (all dimensions were established from the plate center lines).

Openings in shield. The shield is penetrated by numerous openings in the four sides, top, and

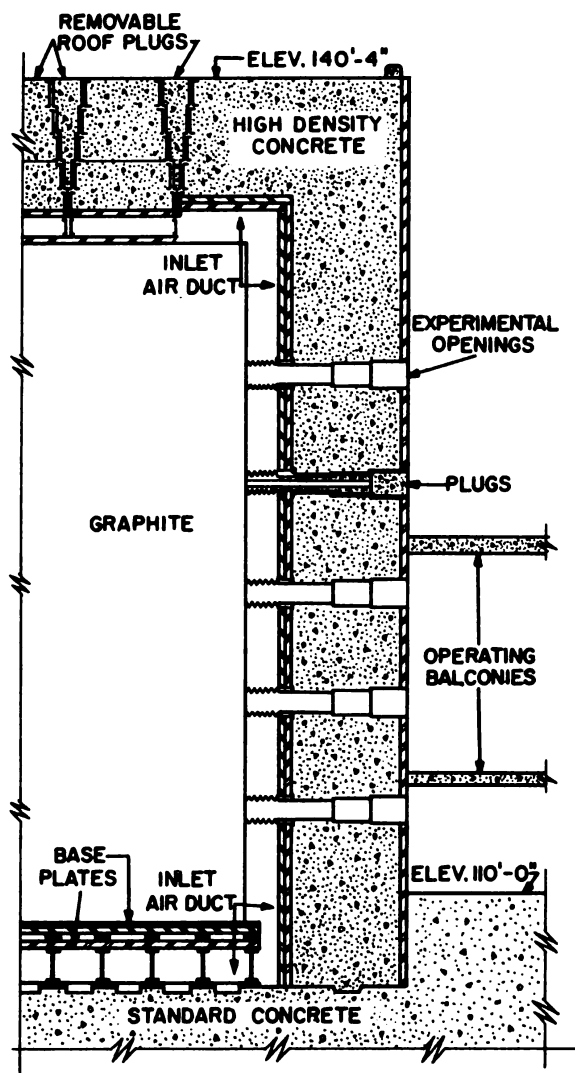


Fig. 6-20 Section through reactor shield.

bottom. These openings vary in size and function but in most cases are alike in being of stepped construction to prevent straight-through radiation leakage. In general they are lined with mild-steel or stainless-steel sleeves extending between the inner and outer 3-in. steel plates and are filled frequently with removable plugs. These plugs are built in several sections and utilize steel, graphite, and high-density grout or concrete. In the experimental openings the air space between the shield and the graphite is bridged by a metal tube; in the rifle barrels, however, there is no connection between the corresponding openings in the graphite and the shield.

There are 14 different types of openings in the shield, as shown in Table 6-8 along with the faces they penetrate.

Table 6-8

Type of opening	Face penetrated
Channel.....	North and south
Scanner.....	Roof
Periscope.....	Roof
Helium trench.....	Roof
Plenum access.....	East and west
Thermocouple.....	Roof
Control rod.....	East and west
Ion (reactor controls).....	East, west, and north
Experimental.....	East and west
Pneumatic tube.....	North
Removable central core.....	North
Shot well.....	Roof
Removable roof.....	Roof
Ion and animal chamber.....	Floor

Figure 6-21 shows a typical rifle-barrel plug.

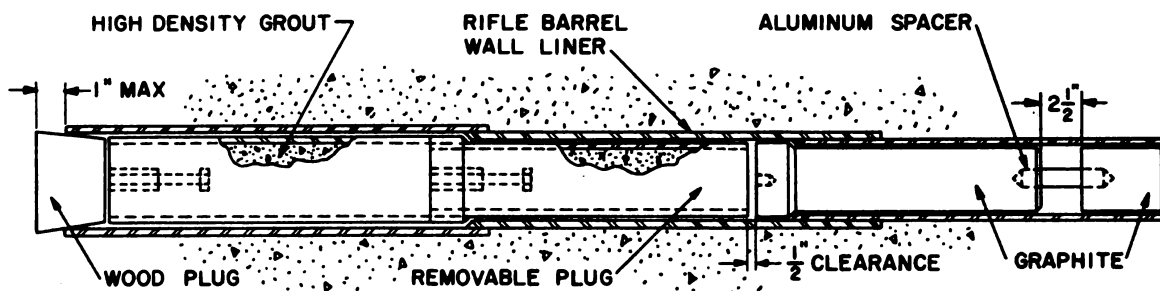


Fig. 6-21 Typical rifle-barrel plug.

COOLING SYSTEMS

The heat generated in the reactor is removed by a stream of filtered atmospheric air which is drawn through the reactor gap and channels, filtered again, cooled, and discharged by centrifugal fans through a stack to the atmosphere.

Primary Air Cooling System. Figure 6-22 is a schematic diagram of the primary cooling system.

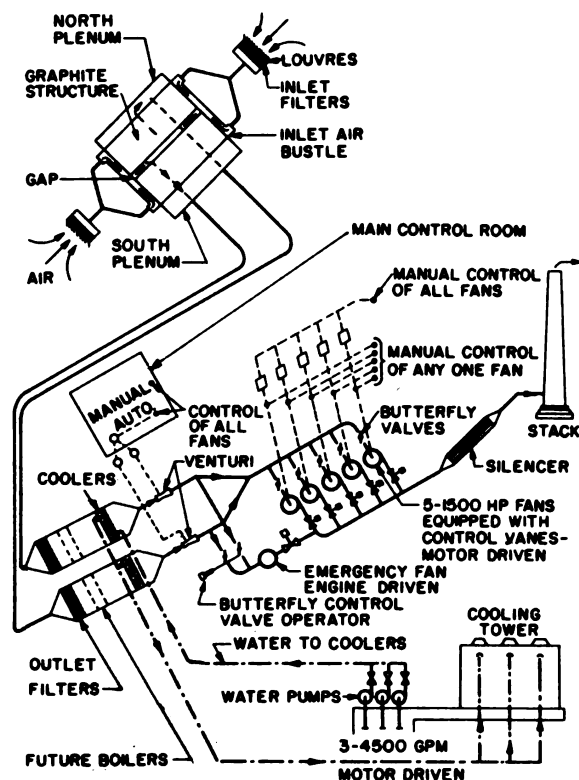


Fig. 6-22 Schematic diagram of cooling system.

The system design was based on the following primary requirements:

1. The air-flow pattern must be bidirectional with intake at the central gap.
2. The power operating level must be 12 watts/cm² of uranium surface at the reactor center.
3. The maximum permissible aluminum temperature can be no higher than 350°C (662°F).
4. Channel spacing must be 8 in. between cen-

ters, and the channel cross section must be 5.6 in.² and round.

5. Air-pressure drop must be no more than 103 mm Hg (57.0 in. H₂O) maximum.

6. The air-entrance design must be a clean slot with discontinuous cartridges.

7. The exhaust air must be cooled with water to 140°F to reduce pumping load.

8. Inlet air filtration must be provided for removal of submicron particles.

9. Exhaust air filtration, suitable for operation at 500°F, must remove particles exceeding 5 μ in diameter.

10. The inlet air flow must be directed over all four split faces of the reactor to keep the concrete shielding from overheating.

11. Air channels must be divided into six groups, with identical restrictions provided in each group, in order to handle the maximum required air flow for each channel.

12. For emergency cooling in case of power failure, a fan must be provided with at least 5 per cent of normal air-flow capacity; the fan should be driven by a gasoline engine.

13. A silencer in the exhaust air duct between fan house and stack must keep the outdoor noise level below 40 db at a distance of 1000 ft from the stack.

Air-flow data. Nominally, the reactor is designed for a maximum power level of 28,000 kw, corresponding to a maximum thermal flux of 12 watts/cm²/sec at the center (when using a 35 \times 35 lattice). For maximum power operation, the calculated air requirement is 1,210,000 lb/hr.

Table 6-9 Calculated Pressure Drops in Air-cooling System (for Core Tube)

Component	Pressure drop, in. H ₂ O
Inlet filters and ducts.....	8.2
Reactor gap.....	9.6
Reactor channels.....	41.1
Exit ducts and filters.....	2.7
Air coolers and venturis.....	3.2
Duct, silencer, and stack.....	1.0
Total.....	65.8

For the sixteen-stage distribution of air flow in the channels, 24 per cent greater flow than the theoretical amount is required, i.e., 1,500,000 lb/hr. Figure 6-23 is a flow diagram of the pri-

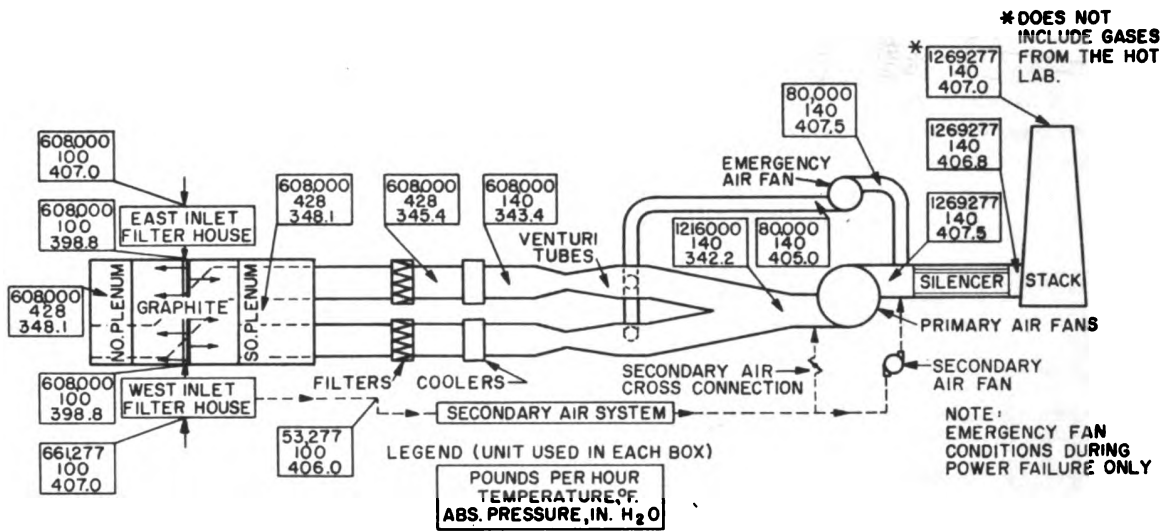


Fig. 6-23 Flow diagram of primary air system.

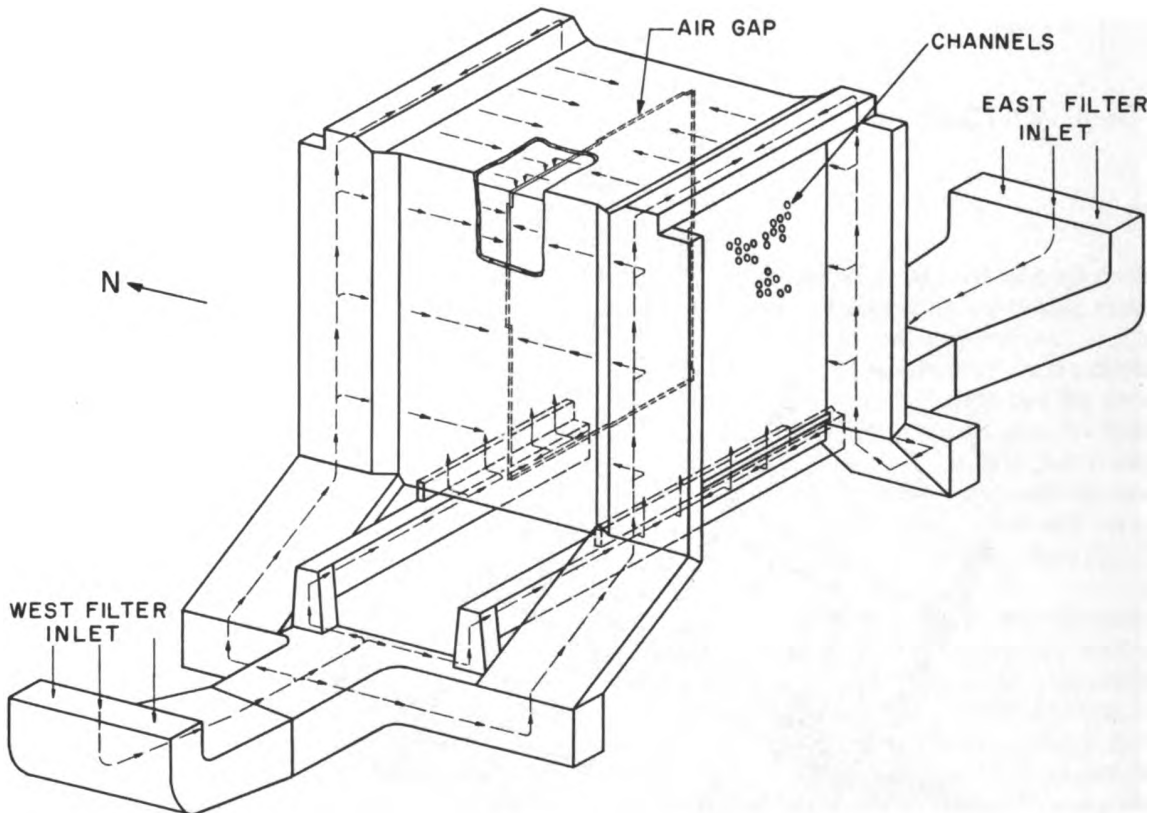


Fig. 6-24 View of inlet air duct system.

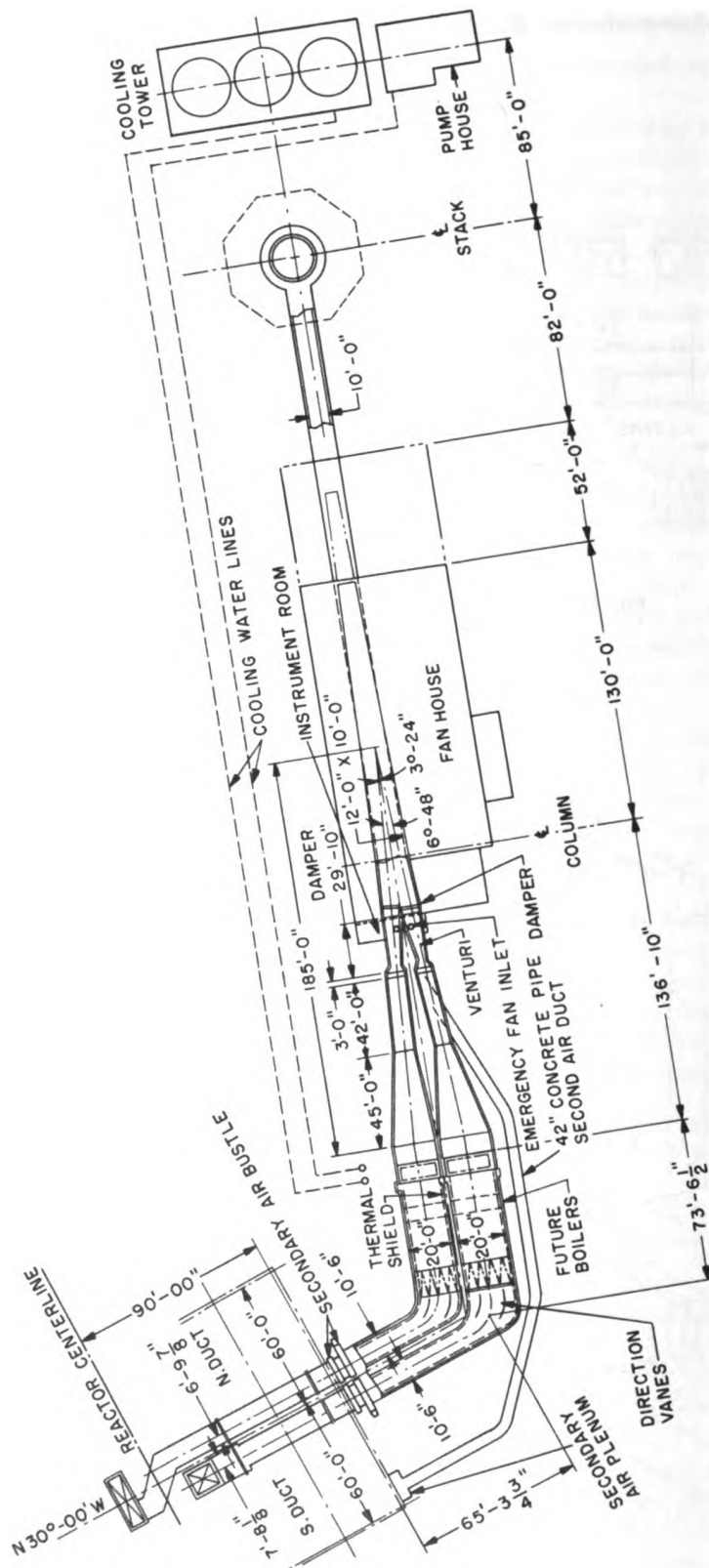


Fig. 6-25 Plan view of outlet duct system.

mary air system for the 28,000-kw power level. Table 6-9 presents a breakdown of calculated pressure drop in the system.

Inlet duct system and filters. Figure 6-24 shows an isometric view of the duct system for the inlet air flow path. Deep-bed pocket filters having an

6-25. Figure 6-26 shows a section through the fan house.

The ducts are rectangular in cross section and are made of concrete. In the case of the interior ducts, this concrete is part of the reactor foundation. The interior surface of the concrete, from

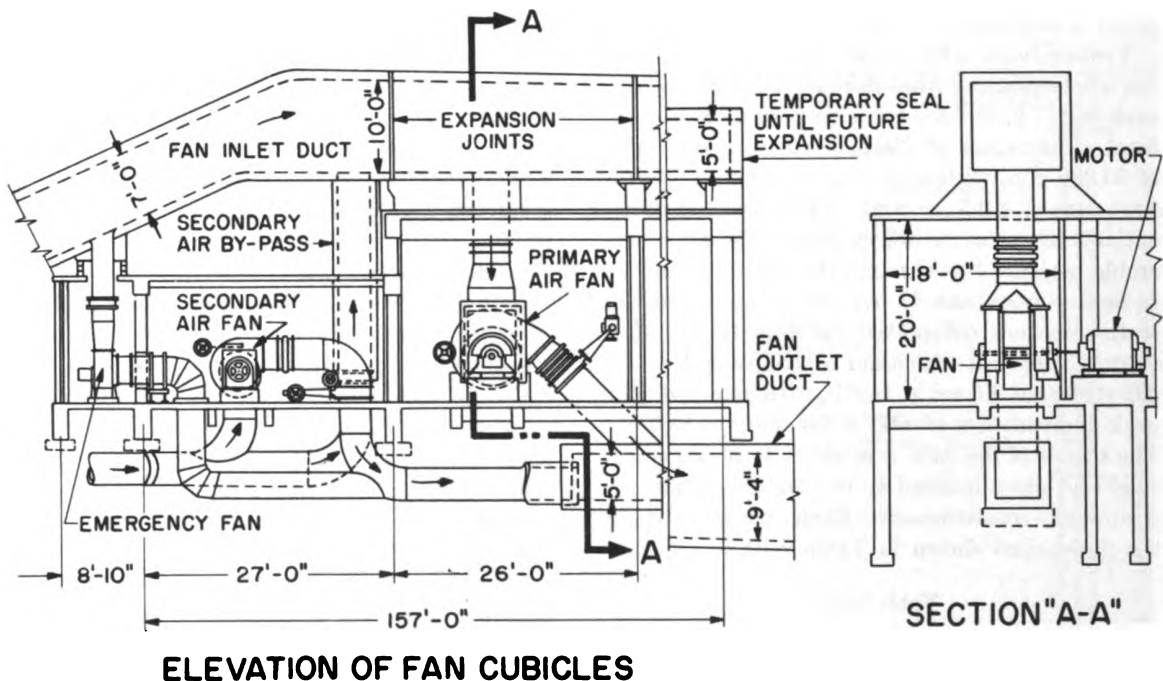


Fig. 6-26 Section through fan house.

efficiency of over 85 per cent are employed in the filter houses.

The reactor gap receives the primary air from the inlet ducts and distributes it to the various channels throughout the reactor. After passing over the hot finned cartridges, the heated air discharges into two plenum chambers, one at each end of the reactor. From both plenums the air passes down into underground ducts, which run southward from the reactor. The plenum chambers are shielded with heavy concrete, the ducts with ordinary concrete.

Outlet duct system. The outlet ducts contain, successively, a filter, cooler, venturi, and damper. The north and south ducts then combine to form a single duct which feeds a battery of fans. These in turn discharge into a duct which contains a silencer and discharges into a stack. A plan of the entire outlet duct system is shown in Fig.

the plenum chambers to the exhaust air coolers, is protected by the secondary air-cooling system, which is described in the next section.

Outlet air filters. The outlet filters are made of a woven glass-fiber cloth guaranteed for service up to 500°F. The filters are designed to remove more than 95 per cent of all particles down to 5 μ and 25 to 30 per cent of smaller particles. When partially loaded, the filters will remove more than 95 per cent of particles down to 3 to 4 μ .

Air coolers. The function of the exhaust air cooler is to cool the hot air leaving the reactor to 140°F, thereby reducing the power requirement for pumping air and the size of the fans required. This results in a saving in investment and operating costs. A total of 1,216,000 lb of air per hour must be cooled from 428°F to 140°F, using water at 85°F as the cooling medium. Two similar sec-

tional tubular heat exchangers are provided, one for each exhaust duct, of the necessary dimensions to permit their installation in the unexpanded ducts. The materials and construction are suitable for operation at 500°F.

The air coolers are supplied with a total of 4800 gpm at an inlet temperature of 85°F. The water is discharged at 120°F.

Venturi tubes. The venturi tubes in the exterior ducts measure total primary air flow through each duct. Each tube has a range of measurement from a maximum of 250,000 cfm to a minimum of 50,000 cfm, with an average error over the entire range of ± 0.5 per cent. The maximum differential static pressure is 8 in. H₂O. The nonrecoverable pressure loss through the tubes is specified to be not more than 13 per cent of the maximum static pressure differential, or 1 in. H₂O. The venturi tube will withstand a collapsing pressure differential of 3.5 psi at 140°F, with an emergency peak temperature of 600°F for short durations. The throat of the tube is made of $\frac{3}{8}$ -in. stainless-steel-clad plate finished to the highest obtainable accuracy of measurement. Each venturi tube has the dimensions shown in Table 6-10.

Table 6-10

Approach duct.....	84 in. square
Rectangular throat.....	84.00 in. high \times 28.96 in. wide
Upstream length.....	5 ft $3\frac{1}{2}$ in.
Throat length.....	15 in.
Downstream length.....	25 ft $3\frac{3}{4}$ in.
Over-all length.....	31 ft $10\frac{1}{4}$ in.

Primary-air fans. The primary-air fans and their associated ducts, valves, and controls are housed in the fan house (Fig. 6-26). A single air duct, into which the two primary-air ducts converge, passes from west to east along the entire length of the fan-house roof and constitutes the plenum from which the five main fans take suction. These fans discharge the air into a duct located below the floor of the fan cubicles in the fan house.

The five main fans are single-stage centrifugal uncooled compressors with a speed of 3580 rpm and capacity of 78,000 cfm at 90 in. H₂O. They are of the differential-pressure type, operating between atmospheric pressures of ± 1 in. H₂O at discharge and 89 in. H₂O at suction, and are powered by 1500-hp 2300-volt induction motors.

The fans are equipped with inlet vanes to effect control of air flow.

The fans are mounted in cells separated from the motor room by either 6-in. concrete walls or 1-in. steel plate to provide necessary shielding for operating personnel. All controls, such as for lubrication, inlet guide vanes, and suction and discharge valves, are available in the motor room.

The instrumentation of the fans permits either automatic or manual control of total air flow or manual control of any particular fan. The quantities of air flowing through each venturi are totalized on a single master flow-recording and controlling instrument, which in turn, through five individual controllers, resets vane positions to maintain the total air flow at a constant predetermined value. The individual controllers are pneumatic instruments designed to divide automatically the load between the fans for the desired air flow. The master controller permits adjustment of the total air flow from the reactor control room. Control of total air flow or manual control of each fan is also possible from the fan house, if desired.

A gasoline-engine-driven fan is provided in the fan house to supply ventilating air for the reactor in the event of power failure. The emergency fan has a speed of 1750 rpm and delivers 20,000 cfm against 2 in. H₂O static head at 140°F. The fan is isolated in a separate cell at the west end of the fan house. A V-belt drive connects the fan shaft extending through the cell wall to an 18.9-hp gasoline engine. An automatic starting circuit is provided to immediately start the engine and open the discharge valve of the fan upon power failure. Another automatic control opens the suction valve of the fan when the main suction-duct vacuum drops to 2 in. H₂O as the main fans slow down after power failure.

Silencer. The silencer, an acoustic filter which reduces the sound level by an amount of 40 db, is installed in the duct connecting the fan house to the stack.

Stack. The stack conveys radioactive fan-discharge air 320 ft above ground level. This permits adequate mixing and dilution with atmospheric air and thus prevents high concentrations of radioactive air from reaching surrounding areas.

Under normal operating conditions the stack will provide a draft of at least 0.2 to 0.3 in. H_2O . In case of fan failure, natural draft with air at $140^\circ F$ is sufficient to ventilate the reactor at 5 per cent of the normal flow. Natural draft can be supplemented by adding live steam; a connection for a steam hose is provided for this purpose in the stack flue.

Secondary Air Cooling System. In order to prevent dehydration of the concrete shielding surrounding the plenum chambers and the primary exhaust ducts, the temperature of the concrete must be kept below $212^\circ F$. This is accomplished by providing an insulating partition, or thermal shield, between the heated primary air and the concrete shield plate (see Fig. 6-27). A secondary

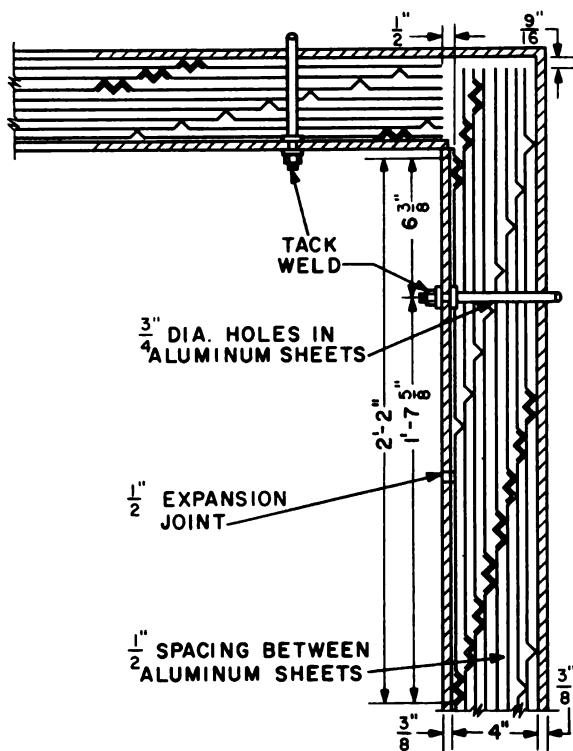


Fig. 6-27 Section of thermal shield.

stream of cooling air is then passed through the space left between the thermal shield and the concrete shield plate. This secondary-air duct is illustrated in Fig. 6-28.

The secondary-air discharge temperature was

planned for $200^\circ F$ maximum on the basis of an assumed inlet temperature of $100^\circ F$. For purposes of design, an average temperature in the secondary air stream of $150^\circ F$ was employed.

Flow circuits. Six individual cooling circuits comprise the secondary-air system (see Fig. 6-29).

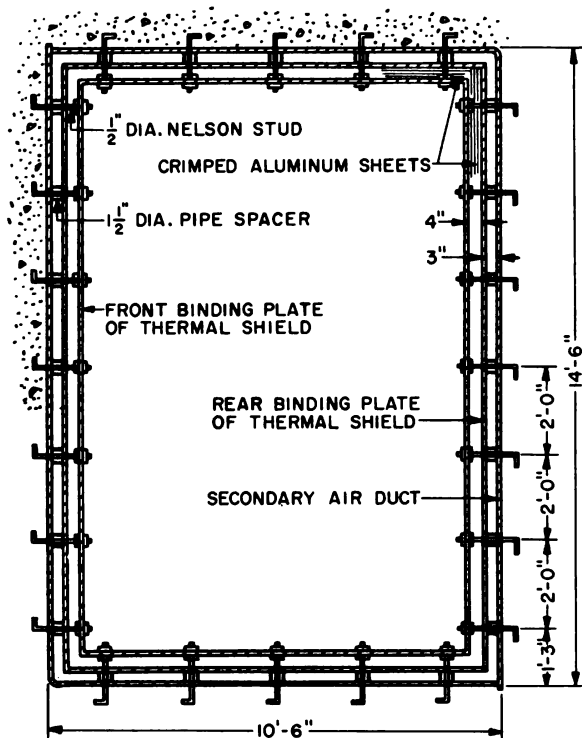


Fig. 6-28 Secondary-air duct.

The air flow through each of these circuits is described below.

The air passes through a filter into a filter box and then into the individual inlet lines. It feeds through a venturi section and then through a butterfly-type manual control valve. At the end of the inlet line the air enters a bustle for distribution to the particular section to be cooled. The air reenters a bustle for collection after leaving the cooling section and is carried by the outlet line to a secondary-air plenum, where the six secondary-air lines join together. The air leaves the secondary-air plenum through a single discharge line, passes through another venturi tube, and then flows to the fan for compression and discharge through the stack.

The sections served by the six cooling circuits and the pipe sizes are given in Table 6-11.

Table 6-11 Secondary-air-system Pipe Sizes

Circuit	Inlet-line pipe sizes, in.		Outlet-line pipe sizes, in.
	In filter house	Filter house to cooling section	
Circuit A, south plenum.....	16	20	20
Circuit B, interior south duct..	14	20	20
Circuit C, north plenum.....	12	16	16
Circuit D, interior north duct..	16	24	24
Circuit E, exterior south duct...	20	20	24
Circuit F, exterior north duct..	16	16	20

Secondary-air fan. The 42-in. common-concrete discharge line connects to the secondary-air fan, which discharges into the primary-air exhaust duct, just ahead of the first main fan. The fan

is located in the west end of the fan house. Fan operating conditions for normal and maximum flow are tabulated in Table 6-12.

Table 6-12

Quantity of air, lb/hr.....	53,300
Suction temperature, °F.....	200
Suction pressure, in. H ₂ O absolute.....	401
Discharge pressure, in. H ₂ O absolute....	408
Differential pressure, in. H ₂ O.....	7
Fan power required, hp.....	25
Size motor installed, hp.....	40
Speed, rpm.....	1170

The secondary-air fan is a single-width single-inlet centrifugal type with overhung rotor and top horizontal discharge. The housing is of welded-steel construction, with a removable cover plate for cleaning. The fan motor is a horizontal constant-speed squirrel-cage induction motor which operates on a 440-volt three-phase 60-cycle current.

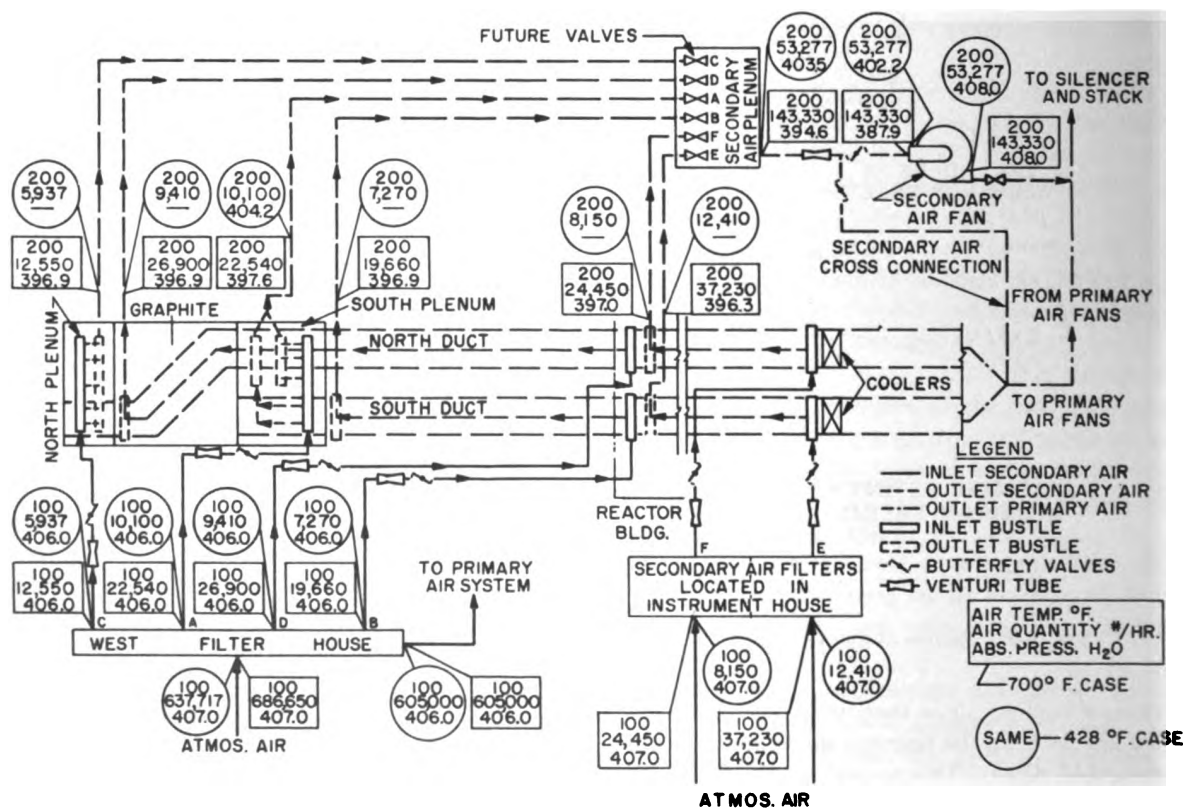


Fig. 6-29 Flow diagram of secondary air system.

CONTROLS AND INSTRUMENTATION

The control system for the nuclear reactor includes 16 motor-driven control rods which are used for emergency shutdowns and in the normal startup, operation, and shutdown of the reactor.

Instrument Groups. The instruments associated with the control and measurement of the reactor power level have been divided into the following five groups:

1. Control rods and drives.
2. Rod-position instruments.
3. Reactor-activity instruments.

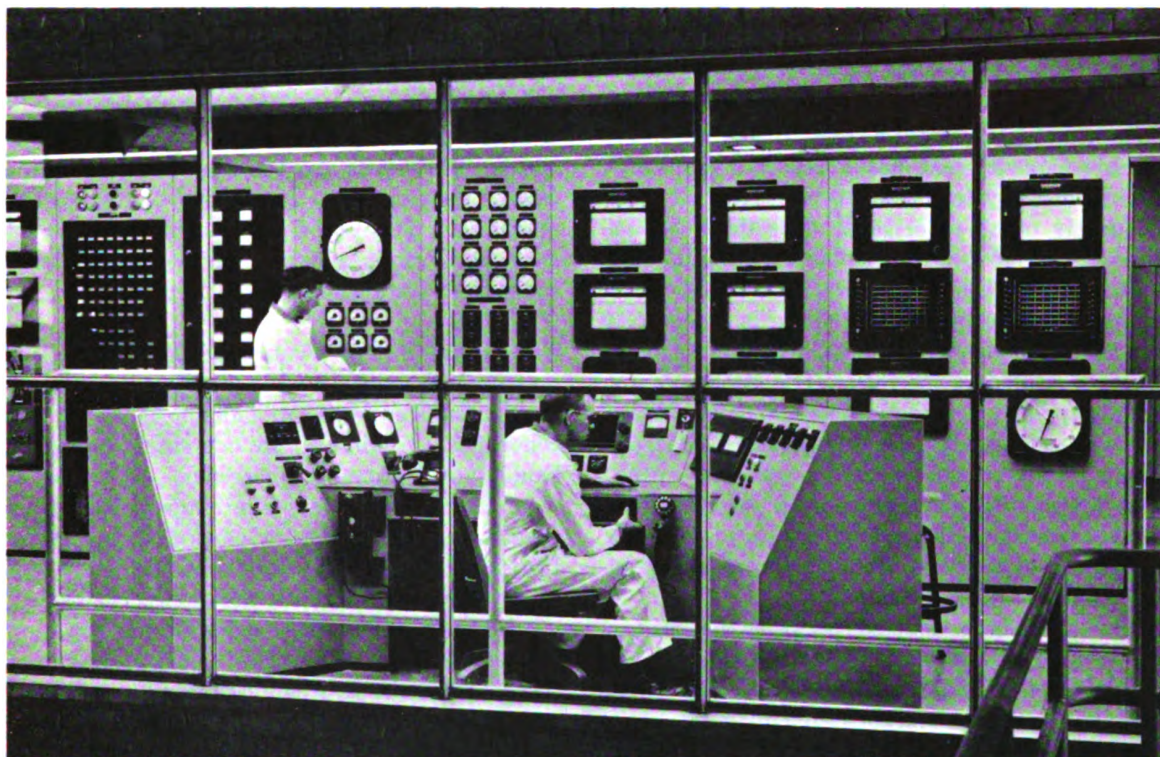


Fig. 6-30 View of control room and operating console.

Furthermore, four boron shot wells are provided to supplement the control rods for emergency shutdown, and an additional emergency safety system introduces a liquid poison (trichlorobenzene) into the pneumatic-tube system of the reactor. A variety of instruments indicate and record the level of the neutron flux in the reactor and the activity of the air in the stack and building. Controls provide protection against excessive level of any of the mentioned measured quantities. The controls and instruments for this system are brought together on an operating console and supervisory control panel located in the control room (see Fig. 6-30). A detailed drawing of the operating console appears in the Appendix.

4. Emergency-shutdown and alarm equipment.
5. Radiation-monitoring equipment.

Control rods. The number and location of the control rods depend upon the excess reactivity required for full-power operation. The excess k requirement for the Brookhaven reactor is 0.021. For rapid shutdown of the reactor, it should be possible to reduce the value of the multiplication factor to 0.99 or even somewhat less. The control-rod system is actually capable of handling an excess k of about 0.035. A maximum of reactor surface for research purposes is made available by averaging the rods along the horizontal diagonals of the reactor face.

The reactor is controlled by means of 16 boron-

steel rods. These contain nominally 1.75 per cent boron and are 2 in. square by 26 ft 9¼ in. long. The control rods extend 25.5 ft into the graphite structure when fully inserted. They penetrate the reactor in directions parallel to the diagonals of the base, entering from the southeast and southwest corners. There are eight rods running in each direction, arranged in two vertical banks of four each. The banks are spaced 4 ft apart on centers and the rods in each vertical bank are 4 ft apart between centers.

Rod drives. Fourteen of the control rods are emergency rods and two are regulating rods. Each emergency rod has a separate electrohydraulic drive system and can be positioned independently of all other rods. The drive system operates as follows: A squirrel-cage induction motor and flywheel drives a variable displacement pump. The output of this pump is piped to a nearby constant-displacement hydraulic motor, which is geared to the rack. The stroke control of the pump is remotely actuated to obtain various rod speeds. In normal operation, the pump is driven continuously at constant speed. The rotation of the hydraulic motor, and therefore the movement of the rod, is controlled in speed and direction by the stroke-control mechanism on the pump.

The flywheel is coupled between the electric motor and the pump and runs at the motor speed of 1765 rpm. If power to the electric motor fails, the stroke control is shifted to FULL-SPEED INWARD, and the kinetic energy in the flywheel is sufficient to drive the rod fully home into the reactor. The flywheel assembly consists of a base casting, two self-aligning sleeve bearings, and a solid-disk flywheel forged integrally with its shaft. The base is supported on the pump-drive base, and the flywheel shaft is directly coupled at one end to the motor shaft and at the other end to the pump shaft.

In the event of an emergency all other control signals are overridden and the emergency rods move to full insertion at the maximum speed of 5.8 fps, traversing the active region of the reactor in 3½ sec. Full inward travel of only one rod is, in general, sufficient for preventing a surge in the power level of the reactor.

The two regulating rods are driven by independent drive units. Two induction motors, a

mechanical differential, and gearing are arranged so that a rod can be moved in or out at high or low speeds of approximately 5 and 0.05 ips, respectively. These rods can be positioned more accurately than the emergency rods. Their primary purpose is to maintain the power level, once it has been established, by removal of the appropriate number of emergency rods. The rods can be operated manually from the operator's console. In addition one of the control rods can be operated from an automatic controller when it is desired that the reactor power be kept constant without operator attendance. The automatic system also assists the operator in reducing the rate of the power rise at the end of the startup operation. Functional diagrams of the emergency-rod system and the automatic regulating-rod system are included in the Appendix.

Additional Safety Systems. The control-rod system will easily control the reactor during startup and shutdown and under normal operating conditions. It will also provide ample control to handle any reasonable emergency. In the event of an accident which may interfere with the proper functioning of the primary control system, additional safety devices are available. These consist of (1) two vertical boron shot wells, (2) two diagonal boron shot wells located in the gap, and (3) a reservoir outside the shield from which trichlorobenzene or a similar liquid can be fed rapidly into the eleven pneumatic tubes.

The gap shot wells are equivalent in effectiveness to at least one control rod, or about 0.005 in k . The vertical shot wells each add 0.001 in k . Thus the four shot wells will provide about 0.007 in k . Filling the pneumatic tubes with trichlorobenzene adds about another 0.01 in k . Thus the combined system is capable of providing almost 2 per cent in k for emergency control. Figure 6-31 shows the assembly of the vertical boron shot well.

Rod-position Instruments. The purpose of the rod-position instruments is to indicate and record the position of the control rods and to provide research data. Four functionally distinct groups of instruments are provided:

1. Position indicators for regulating rods, to a precision of $\pm 1/4$ mm.

2. Position indicators for the emergency rods, to a precision of ± 1 mm.

3. Coarse-rod position indicators for all rods, as an operation aid.

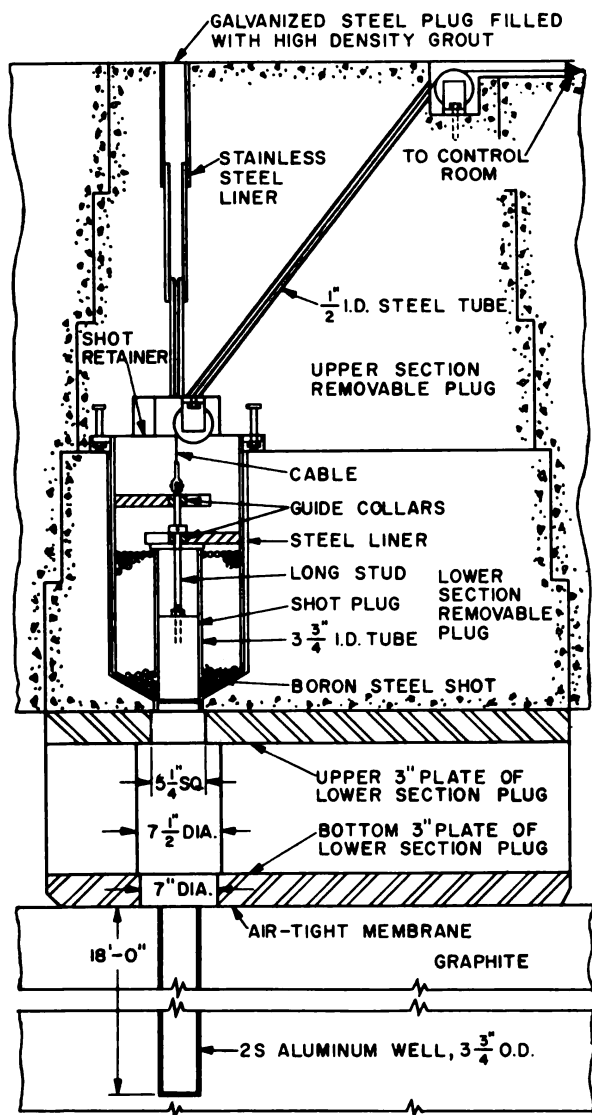


Fig. 6-31 Assembly of vertical boron shot well.

4. Rod-position recorder for all rods to provide graphical records of all 16 rod positions. A diagram of the regulating-rod-position indicating system appears in the Appendix.

Reactor-activity Instrument Systems. Reactor-activity instrument systems are provided to indicate reactor conditions for the guidance of the

operator, who must actuate the alarm and emergency-shutdown systems, and to record work-activity data. A galvanometer system indicates reactor power at high-level operation from full power to 200 watts. A functional diagram of this system appears in the Appendix. The primary element for this system, a graphite differential ionization chamber, is also shown in the Appendix.

A *heat-balance system* is provided which computes the temperature difference across the reactor electrically, multiplies this value by the corrected air-flow value and the specific heat of air, and converts and records this heat balance in terms of kilowatts. Other reactor activity systems may be calibrated against this system.

A *neutron counting-rate system* indicates reactor power at medium and low power levels from shutdowns up to 300 watts. Figure 6-32 shows a

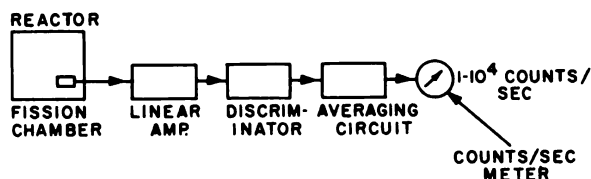


Fig. 6-32 Block diagram of neutron counting-rate system.

block diagram of this system. The fission-chamber primary element for this system is shown in the Appendix. This chamber can be remotely positioned in the reactor by an electric motor which drives a 12-ft lead screw.

A *neutron-counting system* indicates reactor power at the lowest power levels from shutdown to 100 watts. A functional diagram of the system is included in the Appendix. The primary element for this system is a BF_3 , or a boron-lined proportional counter, located in the instrument tunnel under the reactor.

A *trip circuit* actuates the emergency-shutdown system in response to excessive power levels in the reactor and records the power, as measured by an ionization chamber. The circuit comprises three identical sets of equipment which are entirely independent of each other except for their connection to a common recorder. Each set has two separate output channels for the transmission of shutdown signals in response to current from its ionization chamber. One channel is actuated

by Sensitrol relays and the other by an electronic relay. Figure 6-33 shows a block diagram of this

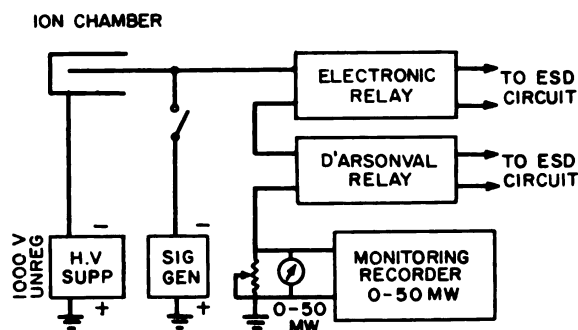


Fig. 6-33 Block diagram of high-level trip circuit.

system. A drawing of the 3-in. boron-coated ionization chamber used in the system is given in the Appendix.

A *logarithmic system* indicates and records the period of reactor power rise, indicates power, and

operates the emergency-shutdown system at selected minimum periods. A functional diagram of this system is included in the Appendix. The graphite differential ionization chamber is identical to that previously shown for the galvanometer system.

A *thermopile system* indicates and records reactor power at high levels and actuates the emergency-shutdown system at excessive levels. Four thermopiles connected to a recorder make up the system. The thermopile element is composed of 42 Chromel-Alumel thermocouples in series. Alternate junctions are boron-coated and are arranged to cancel out the effects of ambient temperature. The complete thermopile is lowered in an aluminum tube 6 in. long and 0.475 in. in diameter.

Additional thermocouple systems are provided for indication and recording of the metal cartridge and graphite temperatures. One hundred metal cartridge temperatures are monitored, and se-

Table 6-13 Emergency-shutdown System

Cause	Shutdown operation	No. of circuits
Power failure of electric motor in emergency-rod drive	Auxiliary contacts in motor contactors open upon loss of 440-volt power, motor overload, short-circuit, or operation of stop push button	14
Loss of auxiliary pressure in emergency-rod hydraulic units	Contacts in pressure switches open when pressure is less than 60 psi (normal pressure, 100 psi)	14
Excessive reactor power	Current in ionization chambers closes Sensitrol-relay contacts which, in turn, open contacts of auxiliary relay	3
Excessive reactor power	Electronic relay removes current from ESD relay coil in response to current in ionization chambers	3
Excessive reactor power	Front-set mercury switch in thermopile recorder opens when thermopile voltage is excessive	1
Excessive rate of increase of reactor power	When current in ion chamber increases too rapidly, output current of logarithmic amplifier causes contacts of Sensitrol relay to close, thus opening contacts of auxiliary relay	1
Excessive metal-cartridge temperature	Front-set mercury switches in single-line recorders open when thermocouple voltage is excessive	2
Loss of cooling-air flow through reactor	Switch in differential-pressure recorder opens, causing auxiliary-relay contacts to open when pressure in either half of reactor is too low	2
Loss of voltage or overload in circuit of ESD solenoids in emergency-rod drives	Insertion of the emergency rods results directly upon loss of voltage to solenoids in the emergency-rod drives; overload or loss of voltage actuates ESD relay by opening a pair of contacts on the air circuit breaker in cabinet in the equipment room	1
Failure of instrument power	Auxiliary contacts on main air circuit breaker in cabinet open upon loss of power or overload	1
Manual operation of push button	Pressing of normally closed push button on console or in fan house causes contacts to open	7
Spare		7

lected thermocouples may be connected to the emergency-shutdown system. Forty-eight graphite temperatures may be monitored.

Emergency-shutdown and Alarm System. Emergency shutdown of the reactor is achieved by inserting all control rods as rapidly as possible. Abnormality in any one of the signaling circuits listed in Table 6-13 will initiate shutdown.

Alarm system. The alarm system is provided to indicate to the operator the specific cause of any shutdown and to warn of abnormal conditions not serious enough to warrant shutdown. A flag-drop-type annunciator is provided for visual and audible notification of these abnormal signals. The upper section of this annunciator contains flag drops which indicate all abnormal conditions that have initiated emergency shutdown, as listed

in Table 6-13. The lower half indicates the abnormal conditions listed in Table 6-14.

Radiation-monitoring Equipment. A *building radiation-monitoring system*, consisting of eight radiation meters located at various stations throughout the building, measures and records the radiation present in the air in the reactor building. The outputs of these meters are connected to the alarm system and recorded in the control room.

An *exit-filter activity system* measures the radioactivity of the dust carried from the reactor by the primary air system. This measurement is taken prior to the filtering of the air and is a continuous monitoring process to assist in deflecting the pressure of radioactive dust or metallic oxide should a break occur in any of the car-

Table 6-14 Alarm System (Signals to the Lower Section of the Annunciator)

Cause	Operation	No. of circuits
Excessive reactor power	A mercury switch in the thermopile recorder opens when the thermopile voltage indicates power is 5% of scale below shutdown threshold	1
Excessive metal temperature	A mercury switch in each of two single-line recorders opens when thermocouple voltage indicates temperature is 5% of scale below shutdown threshold	2
Open circuit in thermocouple	A mercury switch in recorder opens at 2% of scale as pen travels down scale	1
Loss of cooling-air flow through reactor	A mercury switch in the differential-pressure recorder opens, causing auxiliary-relay contacts to open when pressure in either half of reactor is too low	2
Excessive air activity in building . .	A mercury switch at the recorder opens when activity becomes excessive	1
Excessive air activity on exit-duct filters	A mercury switch at the recorder opens	1
Excessive activity in stack air	A mercury switch at the recorder opens	1
Loss of secondary cooling air	A mercury switch in the secondary-air total-flow recorder opens	1
Excessive secondary-air temperature	A mercury switch in the secondary-air temperature recorder opens	1
Excessive primary-air temperature at cooler exit	A mercury switch in the temperature recorder opens	2
Malfunction of fan-room equipment	Switch on fan-room annunciator opens when any flag on that annunciator fails	1
Loss of instrument air pressure	Pressure-operated switch opens when air pressure is too low	1
Failure of 115-volt instrument power	Any of 20 auxiliary contacts on the 20 circuit breakers in cabinet in the equipment room opens when its breaker opens	1
Failure of 440-volt power	Auxiliary contact on undervoltage relay opens when relay drops out	1
Failure of storage-battery charger . .	Auxiliary contact on undercurrent relay opens when relay drops out	1
Fault in helium system	Contacts on helium panel open when recorders indicate fault	2
Malfunction of experimental equipment that may be installed in instrument holes	12
Spare	8

tridges. A flow diagram of this system is included in the Appendix.

A *stack air-activity system* measures the radioactivity of the argon in the air as it enters the stack. The system employs an 8-in. chamber connected to a micromicroammeter. The output of this meter is recorded in the control room and is connected to the alarm system.

LOADING OPERATIONS AND CANAL

The loading and unloading of the various channels in both halves of the reactor are performed from outside the south wall of the reactor structure. The unloading of a cartridge from the north half of the reactor requires the prior removal of the corresponding cartridge in the south half.

The loading elevator is used as the platform from which the loading and unloading is performed. Figure 6-34 shows the south face of the reactor and the loading elevator. Located over the south-plenum roof is a motor-driven jib crane which contains two pneumatic grapples and runs the full length (east-west) of the roof. These hoists are used during loading and unloading operations to hold and position the charging tube and wet storage can in the plenum. The hoists are 7 ft apart and are inserted into the plenum through the two scanner holes in the reactor roof shielding. These scanner holes run the full width of the plenum chamber and are 4 in. wide at the bottom. A number of steps are provided to prevent radiation leakage when the holes are closed with a plug.

In loading and unloading it is necessary to see the plenum. This is accomplished by removing

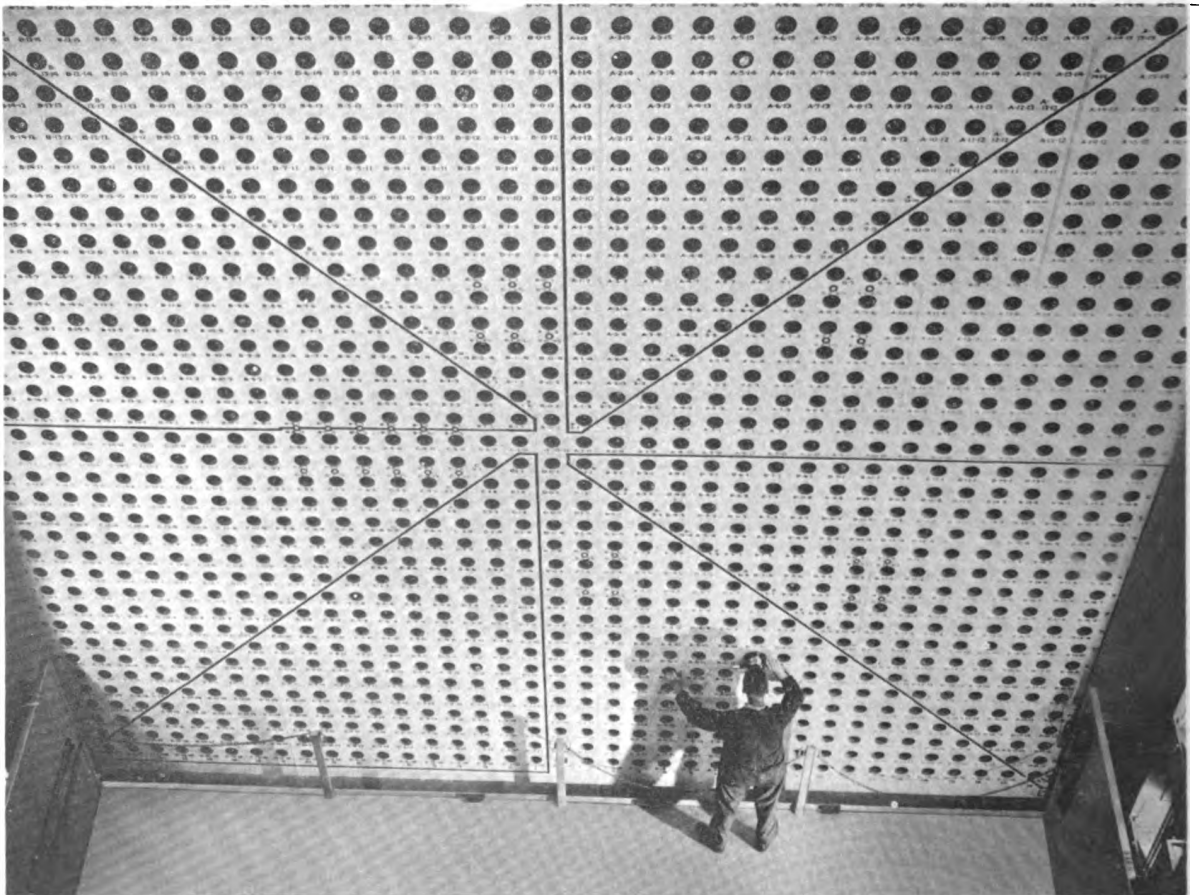


Fig. 6-34 South face of reactor.

the plugs from adjacent rifle barrels, thereby giving a limited view of the area surrounding the channel being loaded or unloaded. In addition to three periscope holes in the plenum roof, the two scanner holes are also used for observation.

3. The loading elevator is brought into position.

4. The two steel-lined high-density grout-plug sections and the inner graphite sections of the rifle barrel are removed and placed in shielded containers.

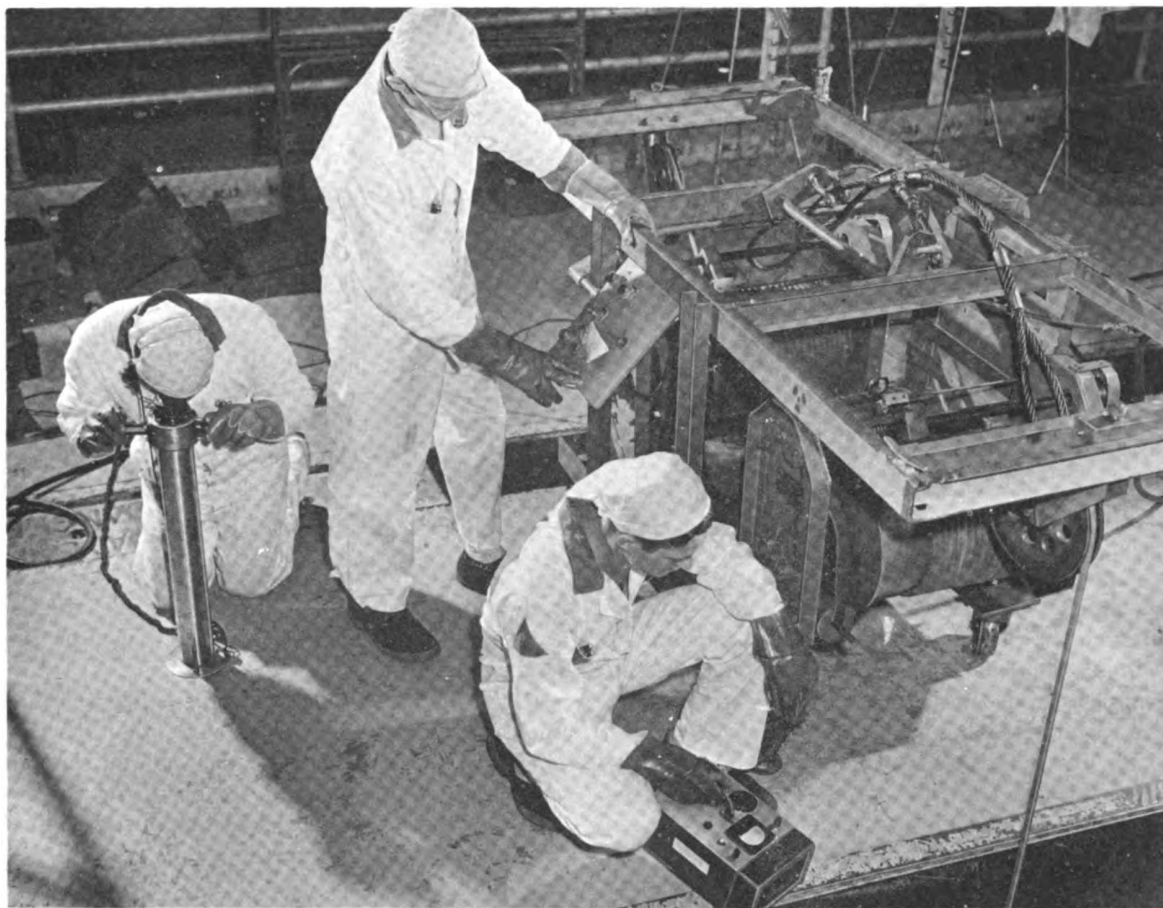


Fig. 6-35 View of jib crane and periscope hole during loading.

The north-plenum roof also contains a scanner hole provided with three periscope openings. Lighting is obtained by the use of drop lights inserted through the scanner holes. Figure 6-35 shows the jib crane and periscope hole in use. Figure 6-36 is a close-up of the south face of the reactor during a loading operation.

Unloading Procedure. The steps involved in unloading a south cartridge are as follows:

1. The two scanner plugs are removed.
2. The jib crane is positioned over the holes.

5. Adjacent rifle-barrel plugs are removed for additional visual aid.

6. The periscope plugs in the plenum roof are removed for visual aid.

7. The wet storage can is positioned on the elevator. (This is a 2 $\frac{7}{8}$ -in. OD steel can 11 ft 8 $\frac{1}{2}$ in. long with $\frac{3}{32}$ -in. walls. The north end of the can is open. The south end of the can contains a cartridge-limit stop pin. A sleeve 8 ft 8 $\frac{1}{4}$ in. long with a 2 $\frac{15}{16}$ -in. ID is fitted over the south end of the storage can for a distance of 1 ft 8 $\frac{1}{4}$ in. and acts as a bridging tube.)

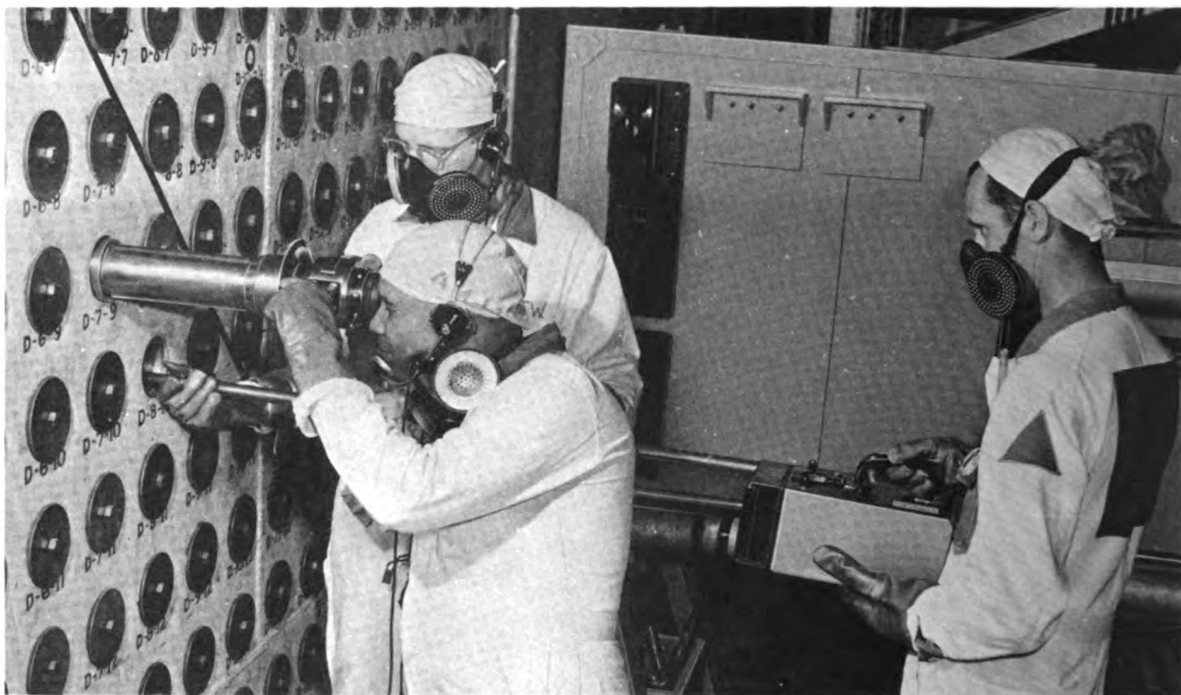


Fig. 6-36 Close-up of south face during loading operation.

8. The sleeve is placed over the south end of the storage can and the combined unit is pushed through the rifle barrel into the south plenum.

9. As the can moves into the plenum, the grapplers from the hoists are placed around the can to support it.

10. The outer hole plug is removed by inserting tools through the rifle barrel and can. On removal these items drop to the plenum floor.

11. A cutting tool is inserted as above and the helium line is snipped.

12. The helium line is disconnected from the manifold and removed.

13. The cartridge-removal tool is inserted beneath the cartridge fins, and the cartridge is pulled into the storage can until it hits the limit stop pin and then the tool is removed.

14. The can is lowered by the hoists and movement of the crane over the discharge chute.

15. The storage can is lowered down the incline and into the deep pit and canal.

The unloading procedure for a north cartridge is very similar.

Loading Procedure. The procedure for loading a north cartridge into the reactor is as follows:

1. The aluminum cartridge with attached helium line (contained in a seamless steel storage tube) is brought from storage and placed on the loading elevator.

2. An aluminum charging tube is inserted into the rifle barrel passing through the concrete shielding. Across the south plenum and through the corresponding opening, the north end is positioned in the tapered outlet section of the channel (in the south half of the reactor).

3. A flexible electrician's "snake" is inserted into the proper stainless-steel conduit of the helium system until it emerges at the end of the south channel.

4. The snake is connected to the end of the helium line on the cartridge.

5. The snake is removed from the plenum tube, pulling the helium line into the charging tube across the plenum and through the channels.

6. When the snake has been removed, the cartridge is pushed through the charging tube, using a 1-in.-diameter aluminum pushing rod.

7. After the cartridge has crossed the gap and finally drops into the anchor slot of the north half of the reactor, the pushing rod is removed.

8. The helium line is connected to the manifold.

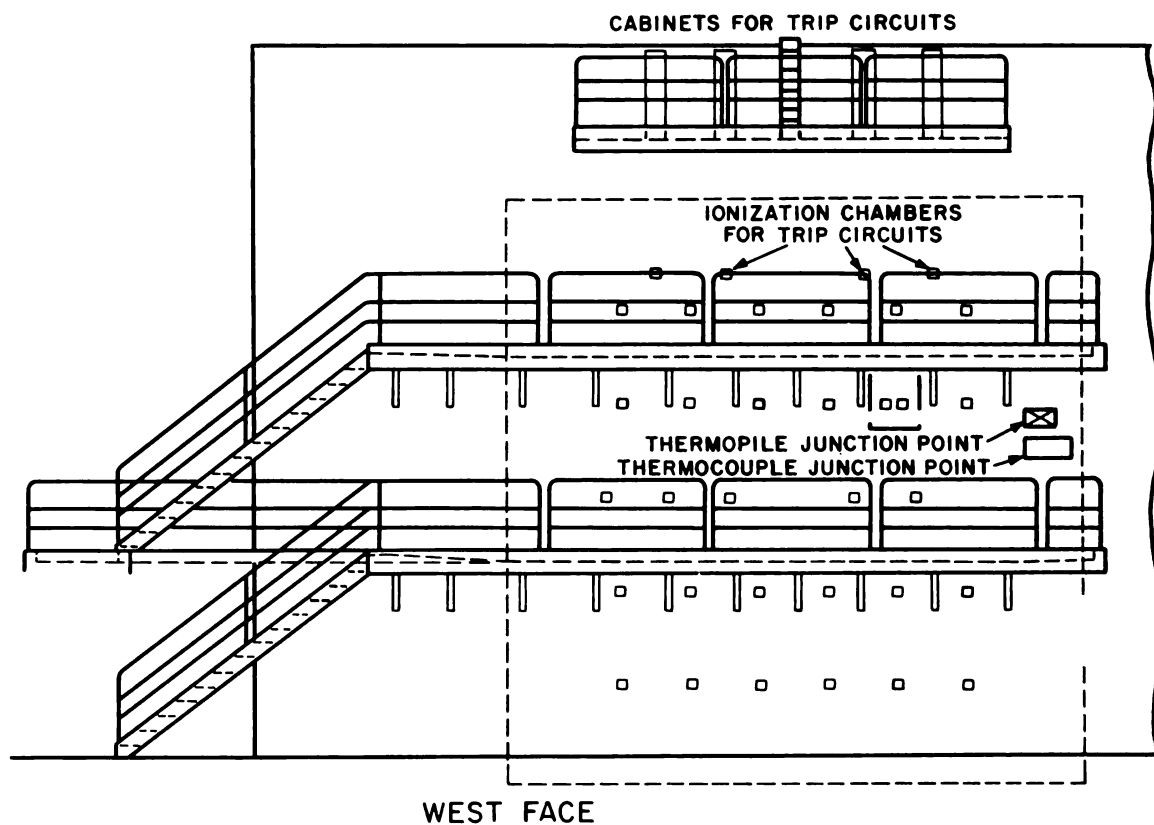
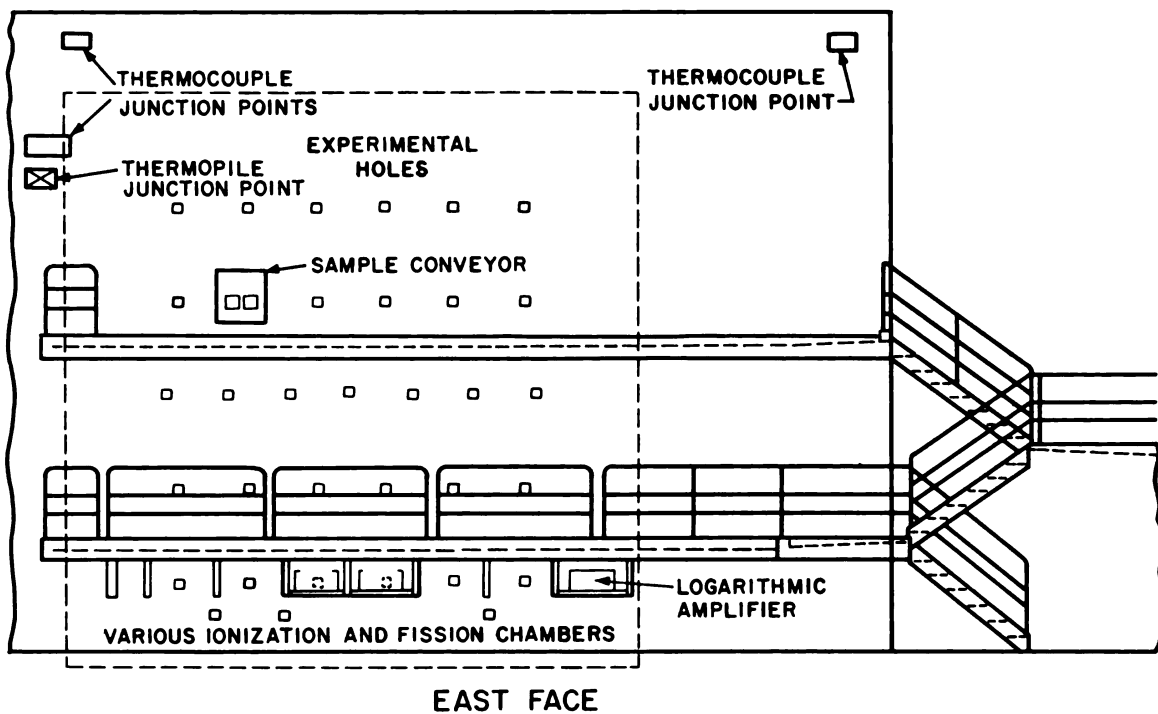


Fig. 6-38 Location of experimental facilities.

Experimental Holes. Thirty experimental holes, each 4 by 4 in. in cross section, extend through the reactor from the east to the west faces. The holes form a square matrix 15 ft on each side. When not in use, the holes are filled with grade AA graphite plugs, 3.960 in. square and 12 to 30

When not in use, the gap is filled with a core of graphite blocks, which runs continuously for 25 ft. The core utilizes bars 3.960 in. square, identical with those used to plug the experimental holes. The access opening for the removable core penetrates the north shield only.

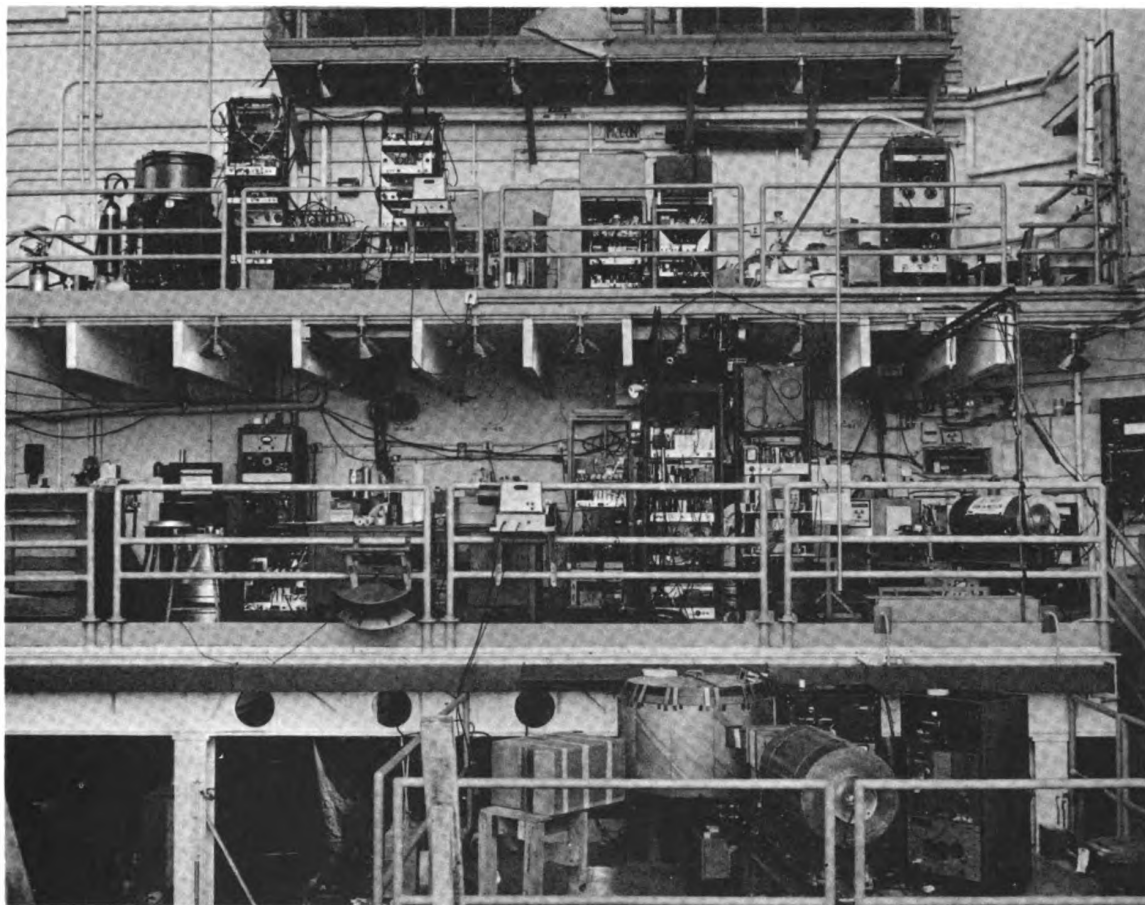


Fig. 6-39 Recent photograph of west face of reactor showing experimental facilities.

in. long. Each block is tapped at both ends with a 1-in. standard-threaded hole. These holes accommodate a threaded steel rod used in inserting and removing the bars. Figure 6-39 is a recent photograph of the west face of the reactor, indicating the usage of experimental holes.

Removable Central Core. A 12- by 12-in. opening, which runs along the north-south axis of the reactor, is provided for general experimental use and also for thermal-column experiments.

Sample Conveyor. The *sample conveyor* is an arrangement for exposing a large number of small samples to radiation for a predetermined length of time. Essentially it consists of an endless train of about 430 graphite sample holders which are pushed through the reactor in an east-west and west-east direction by a reciprocating mechanism. The general arrangement of the sample conveyor is illustrated in Fig. 6-40. Figure 6-41 shows an operator sighting through the periscope of the sample conveyor.

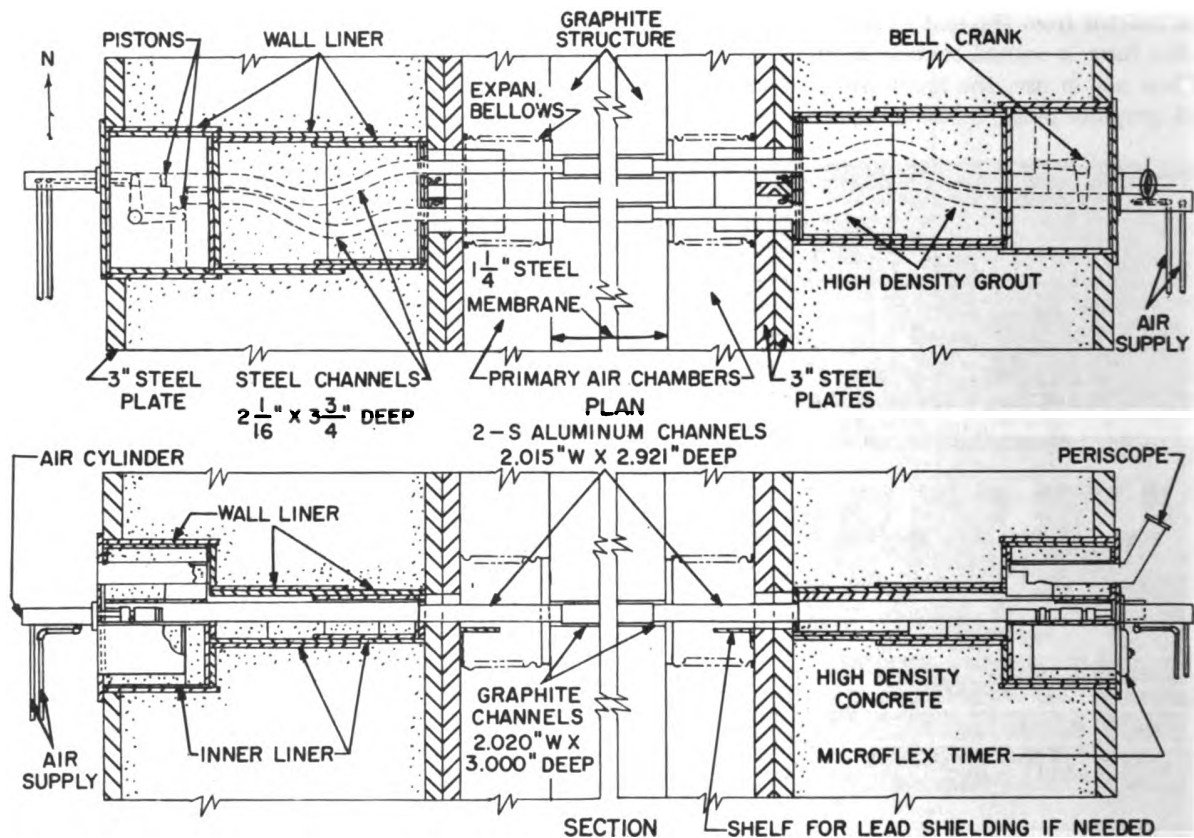


Fig. 6-40 General arrangement of sample conveyor.

Pneumatic Tubes. The pneumatic-tube system provides a means for rapidly inserting samples into the reactor and removing them. In this way samples may be subjected to high flux for short, controllable periods, and the irradiated samples of short half-life may then be conveyed to the point of use as quickly as possible. The easily operated installation minimizes direct handling and radiation hazard. The system permits transfer of a specimen from the reactor center to the face in 0.25 sec, transfer to the reactor laboratory in 5 sec, and to the hot laboratory in 9 sec. A schematic diagram of the pneumatic system is given in Fig. 6-42. The system is also illustrated in Fig. 6-43, which shows the north face of the reactor.

A 20- by 20-ft section of the top shield is removable in 4-ft squares by means of the overhead crane. These openings provide for experimental

tion requiring relatively large areas of radiation exposure, such as thermal column and shielding experiments. The removable roof section is composed of an interlocking arrangement of stepped plugs to prevent radiation leakage through cracks. A total depth of 6½ ft of shielding is arranged in five steps. Part of the experimental area on top of the reactor is illustrated in Fig. 6-44.

The removable section is centered over the graphite structure and covers an opening 20 ft 1 in. square at the bottom and 21 ft 3 in. square at the top. The plugs are made in two layers with the lower layer 3 ft 6 in. deep and the upper layer 3 ft deep.

Animal and Instrument Tunnels. The experimental openings in the graphite structure are limited in size by lattice requirements. In order to expose large samples to radioactivity, two



Fig. 6-41 Operator sighting through periscope of sample conveyor.

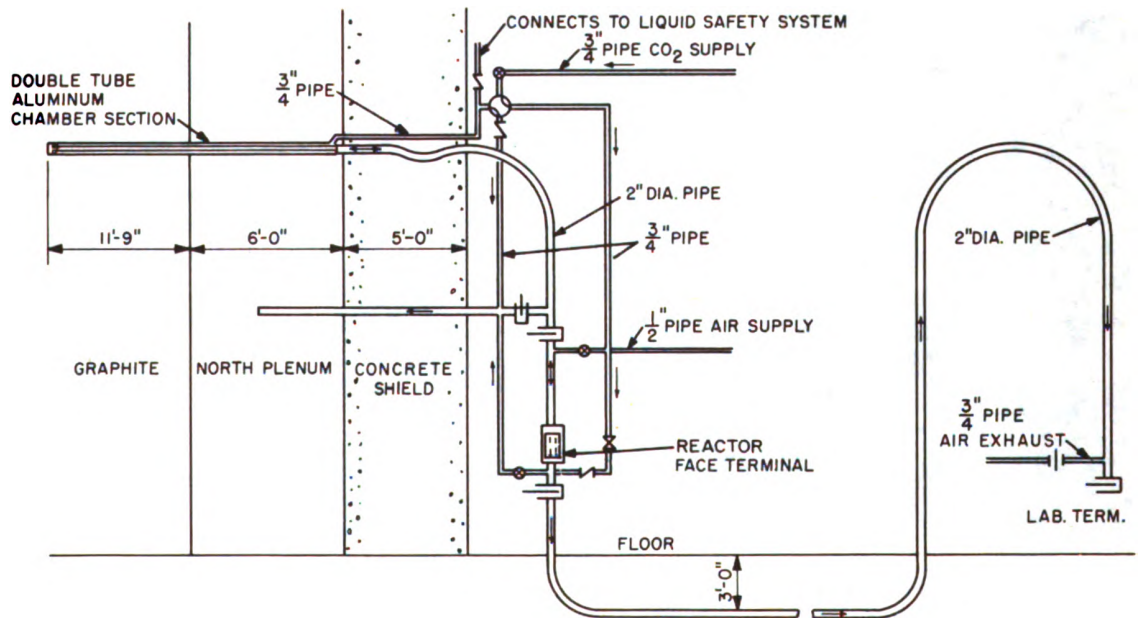


Fig. 6-42 Schematic diagram of pneumatic-tube system.

chambers have been provided underneath the reactor. One of these, the animal chamber, is employed to irradiate animals (see Fig. 6-45). The other, the instrument chamber, is employed to irradiate miscellaneous large samples and is serv-

iceable in a variety of ways. The two chambers are located in the layer of concrete (5 ft 10 $\frac{1}{4}$ in. thick) between the top of the buttresses and the bottom of the I beams supporting the reactor.

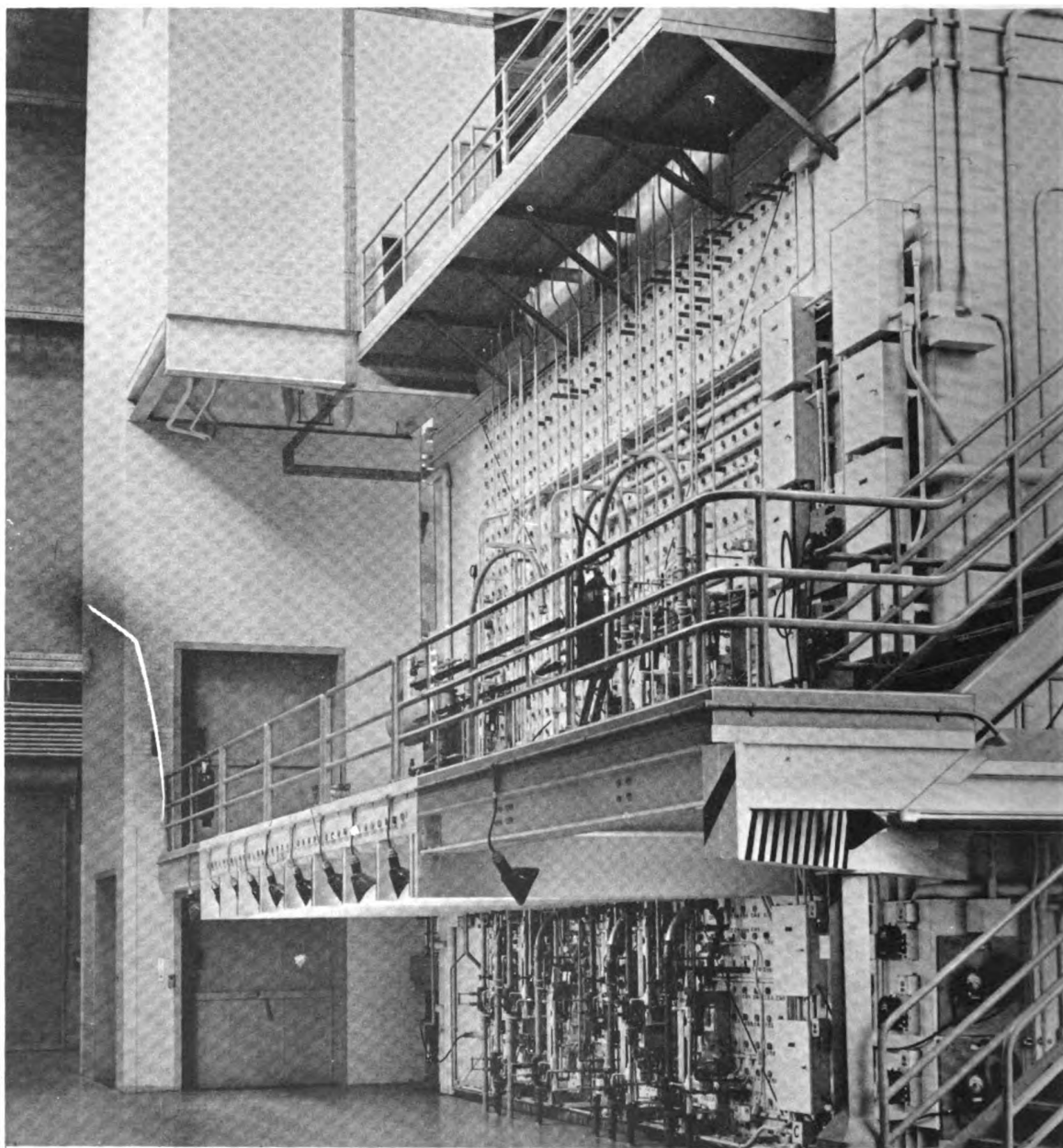


Fig. 6-43 View of north face of reactor showing pneumatic-tube system.

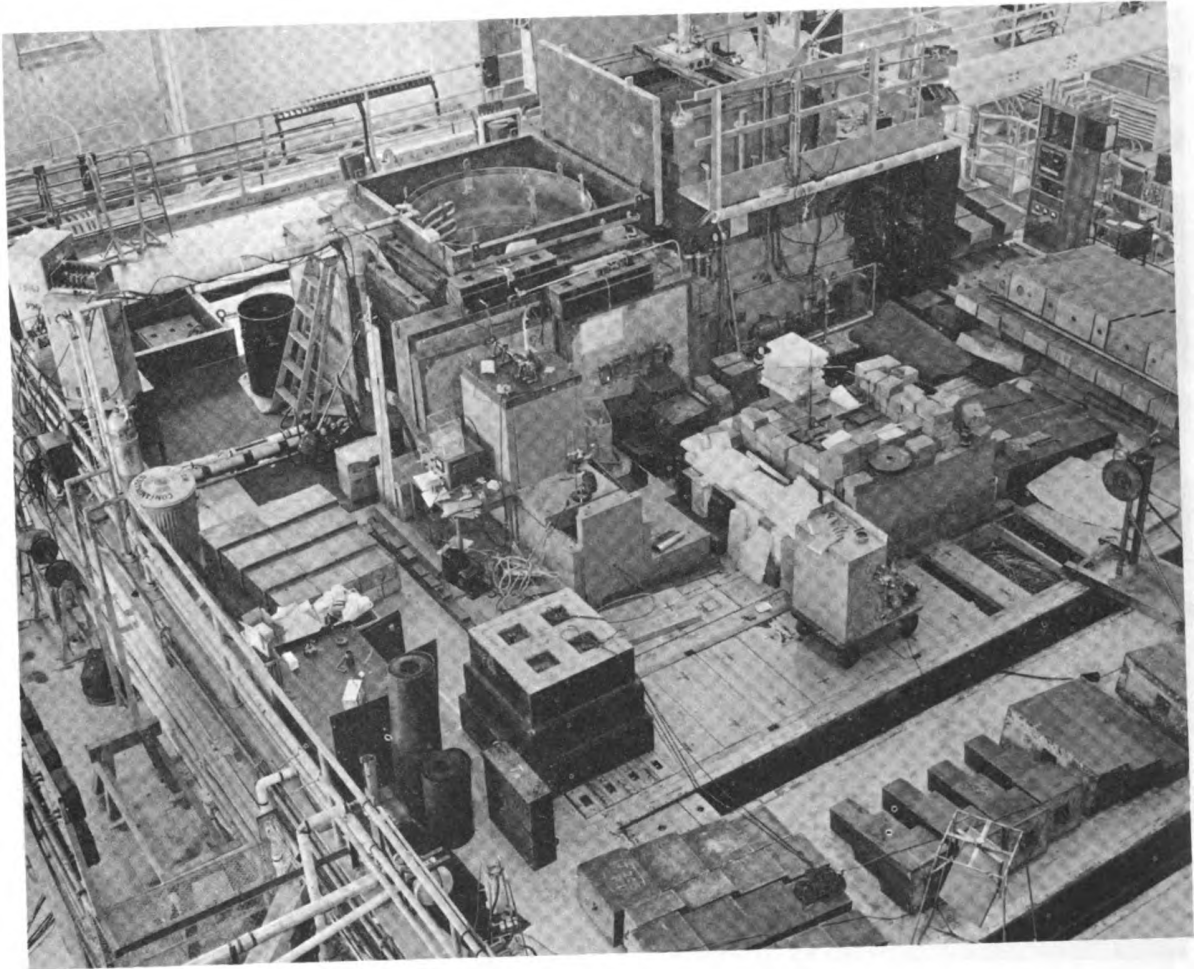


Fig. 6-44 View of experimental area on top of the reactor.

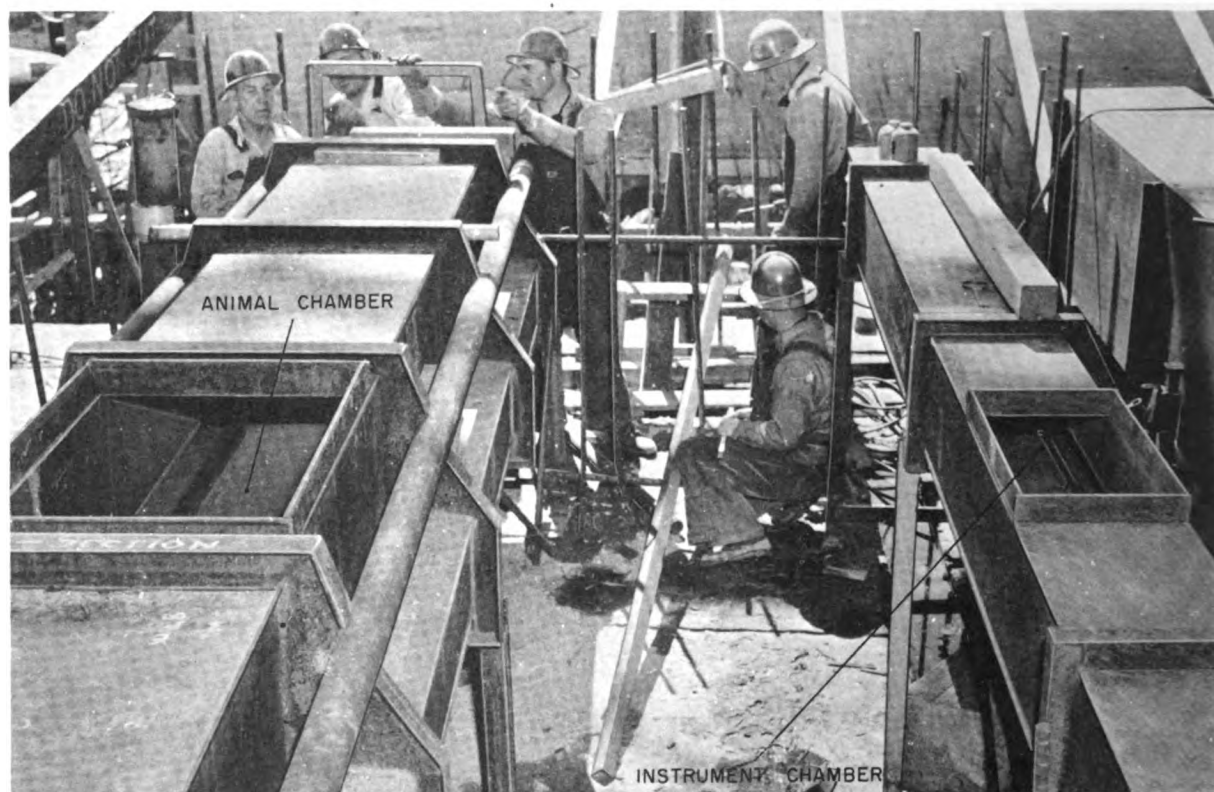


Fig. 6-45 Animal and instrument chambers during construction.

APPENDIX

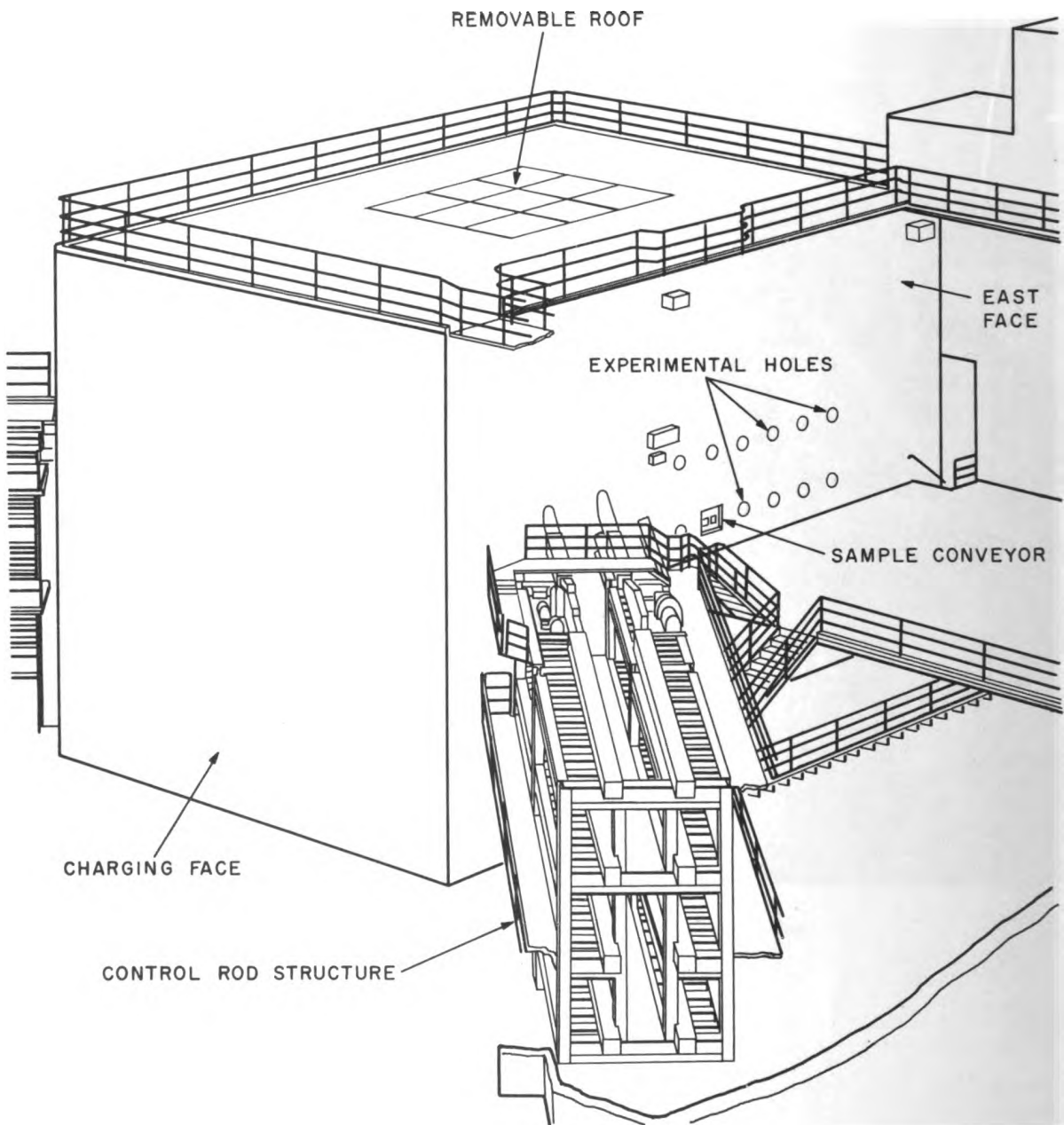


Fig. 6-46 Isometric view of reactor, showing location of control rods and experimental holes.

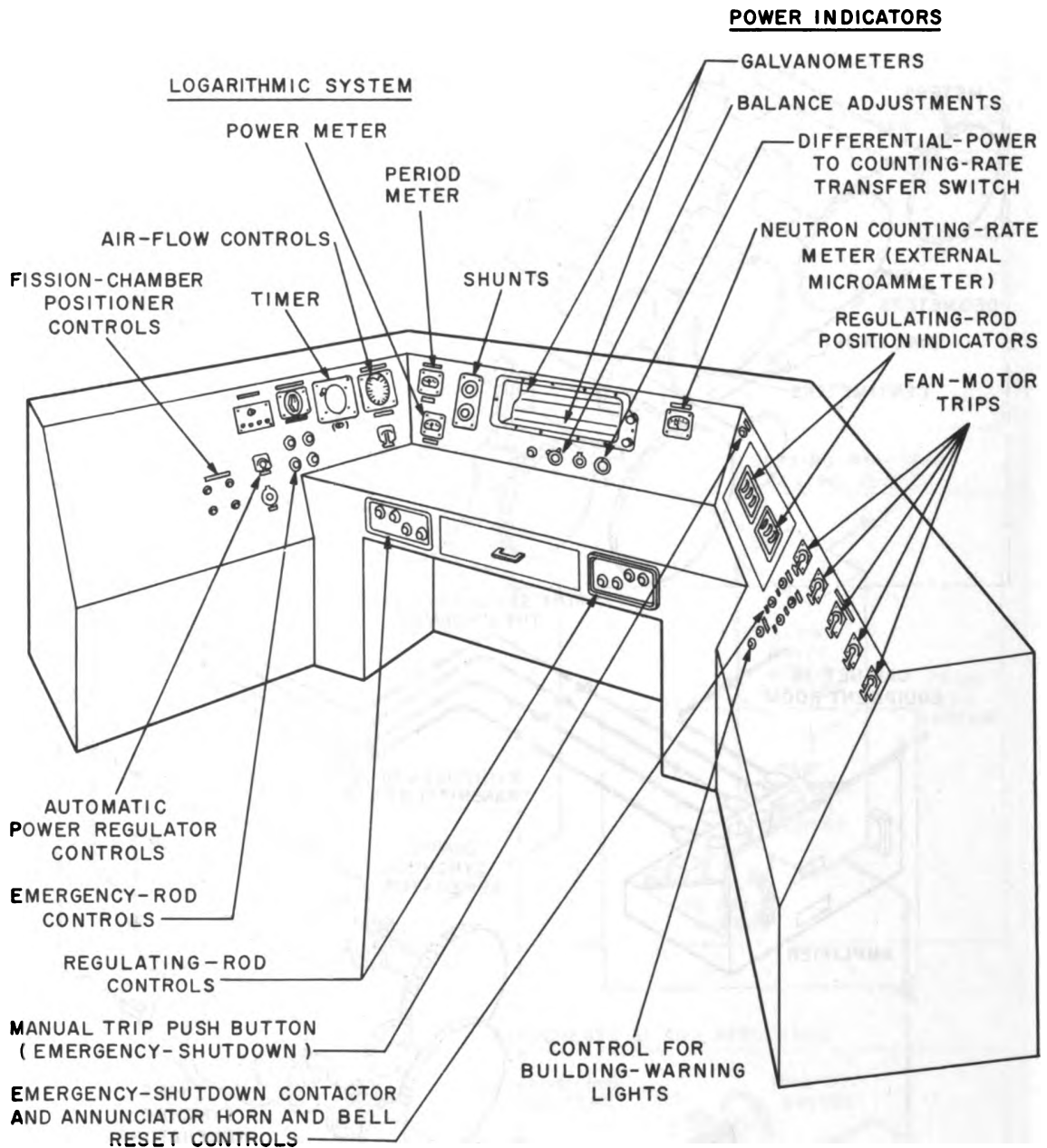


Fig. 6-47 View of operating console.

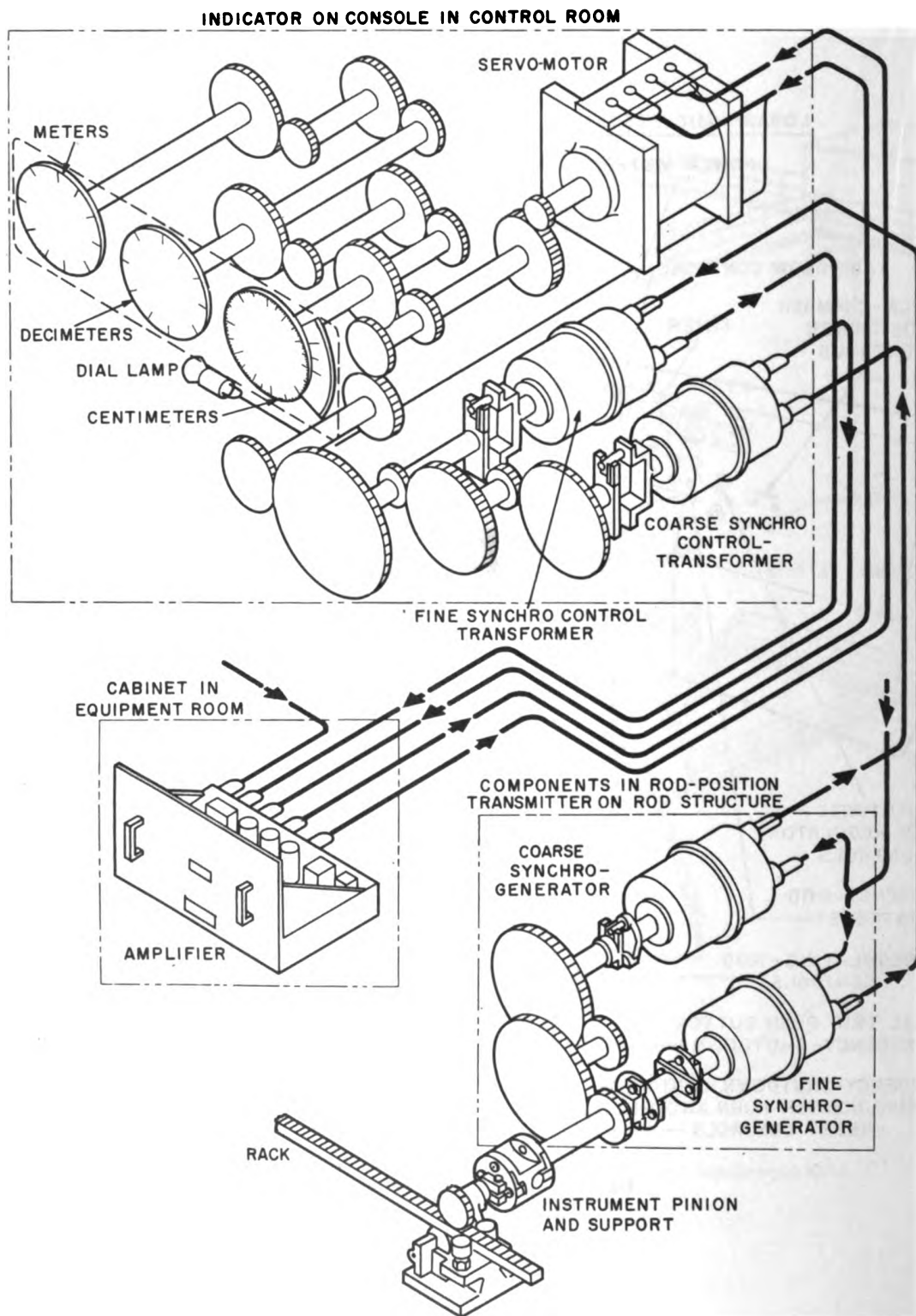


Fig. 6-48 Regulating-rod-position indicating system.

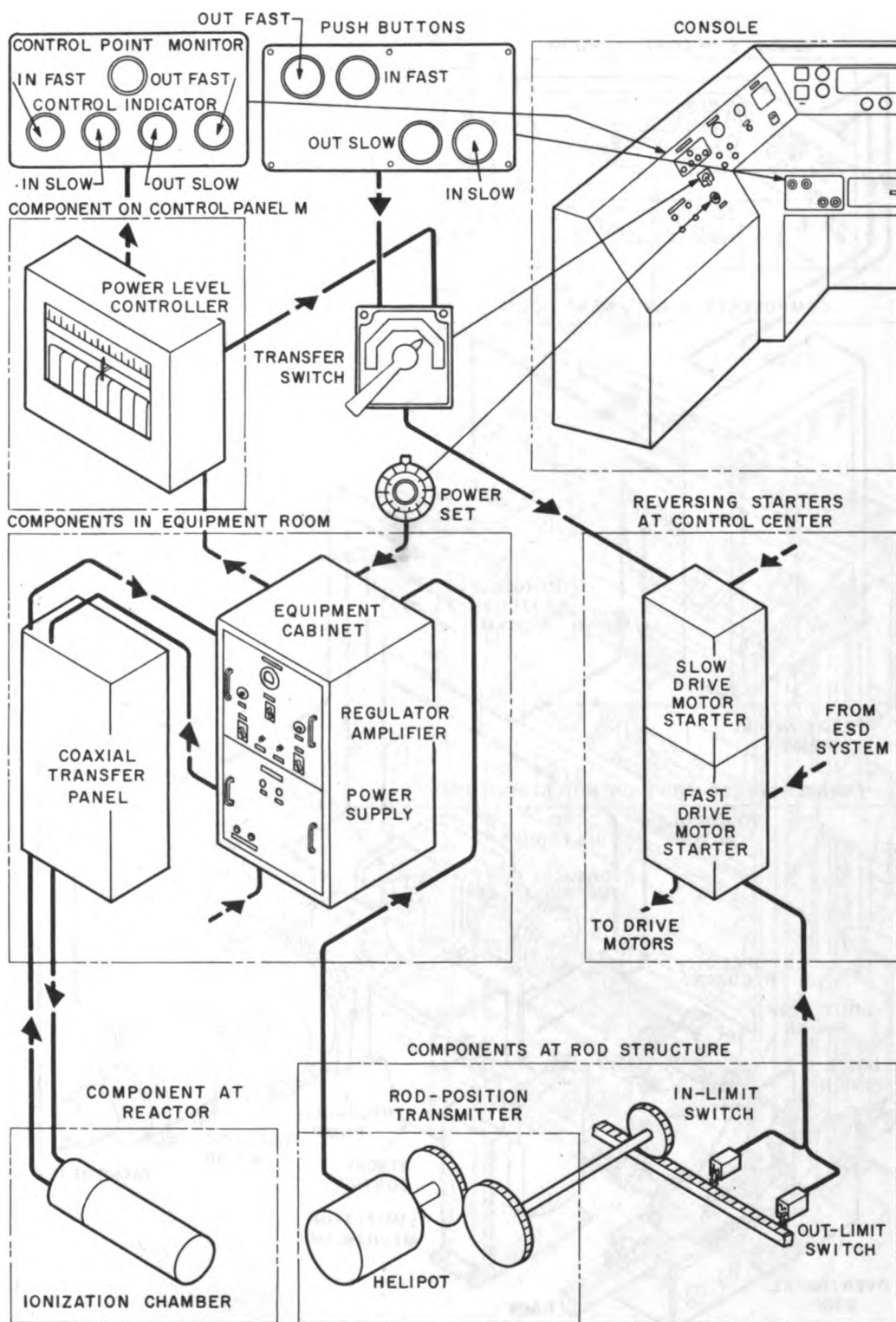


Fig. 6-49 Regulating-rod system, rod No. 15.

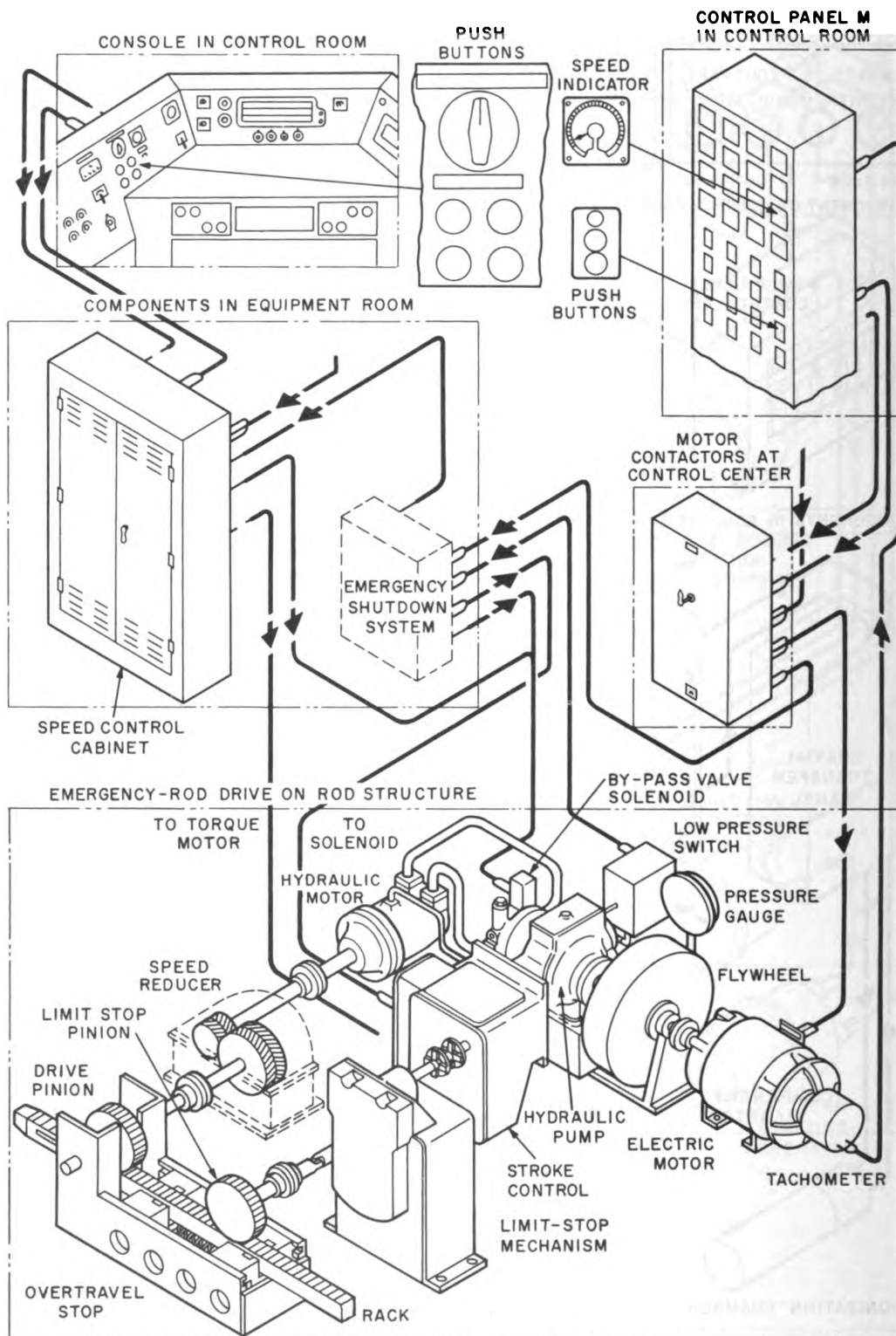


Fig. 6-50 Emergency-rod system.

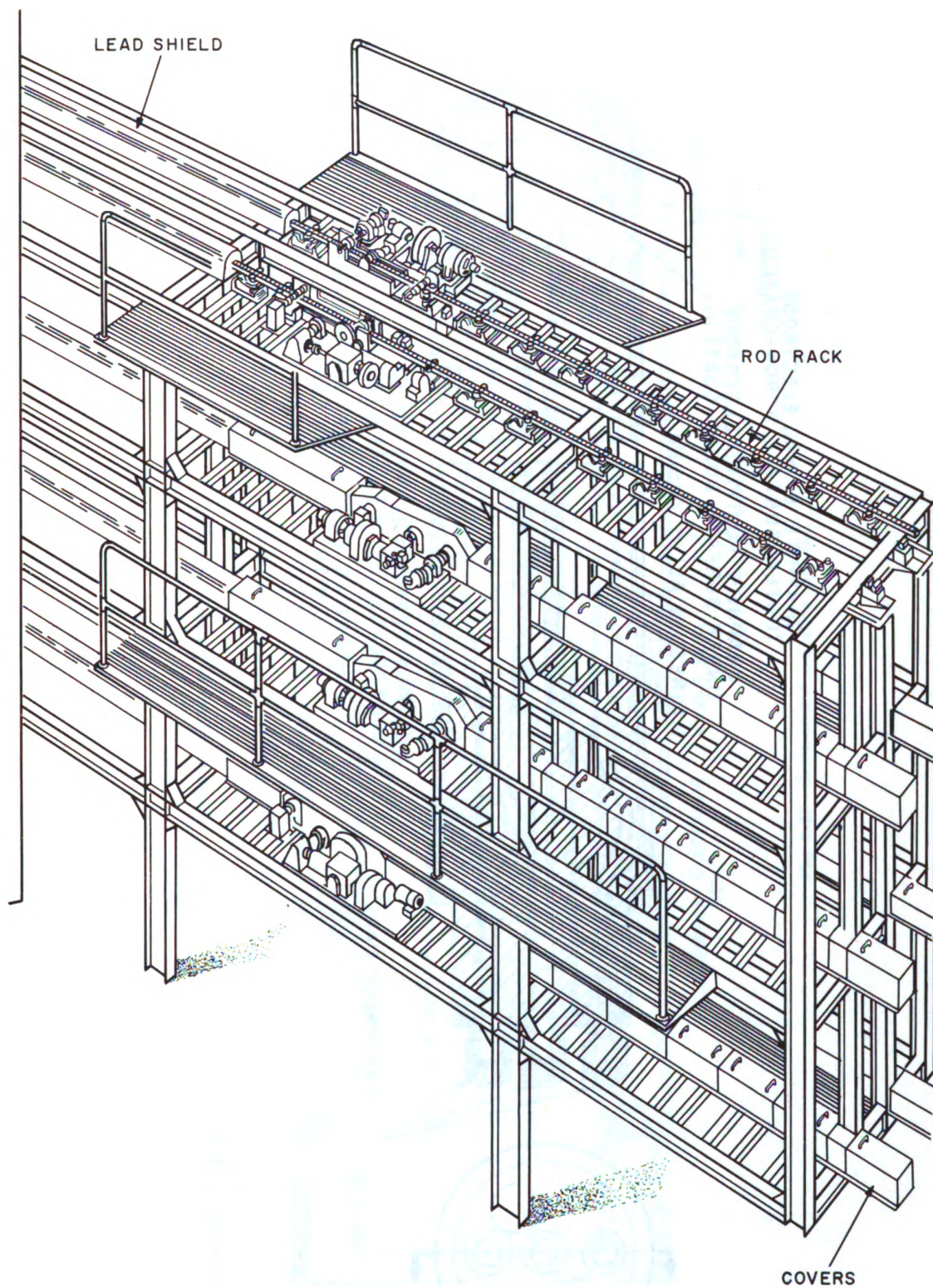


Fig. 6-51 Equipment arrangement on control-rod structure.

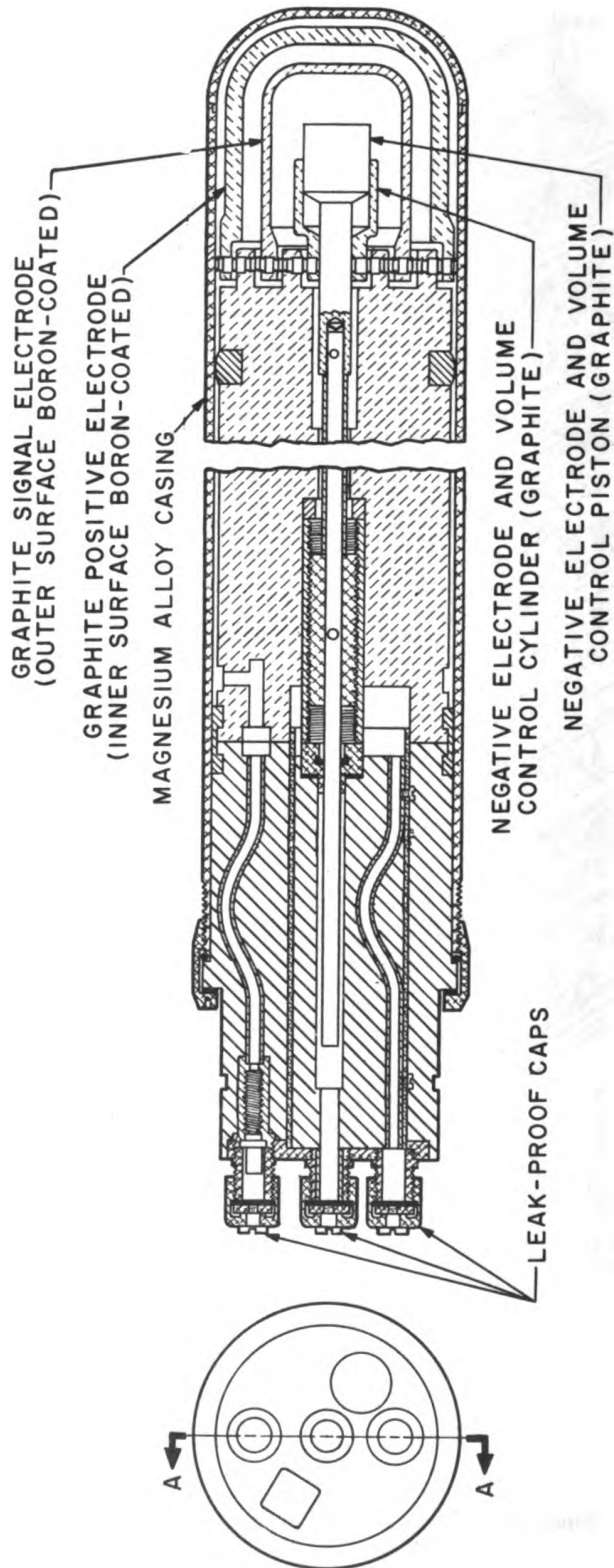


Fig. 6-52 Graphite differential ionization chamber.

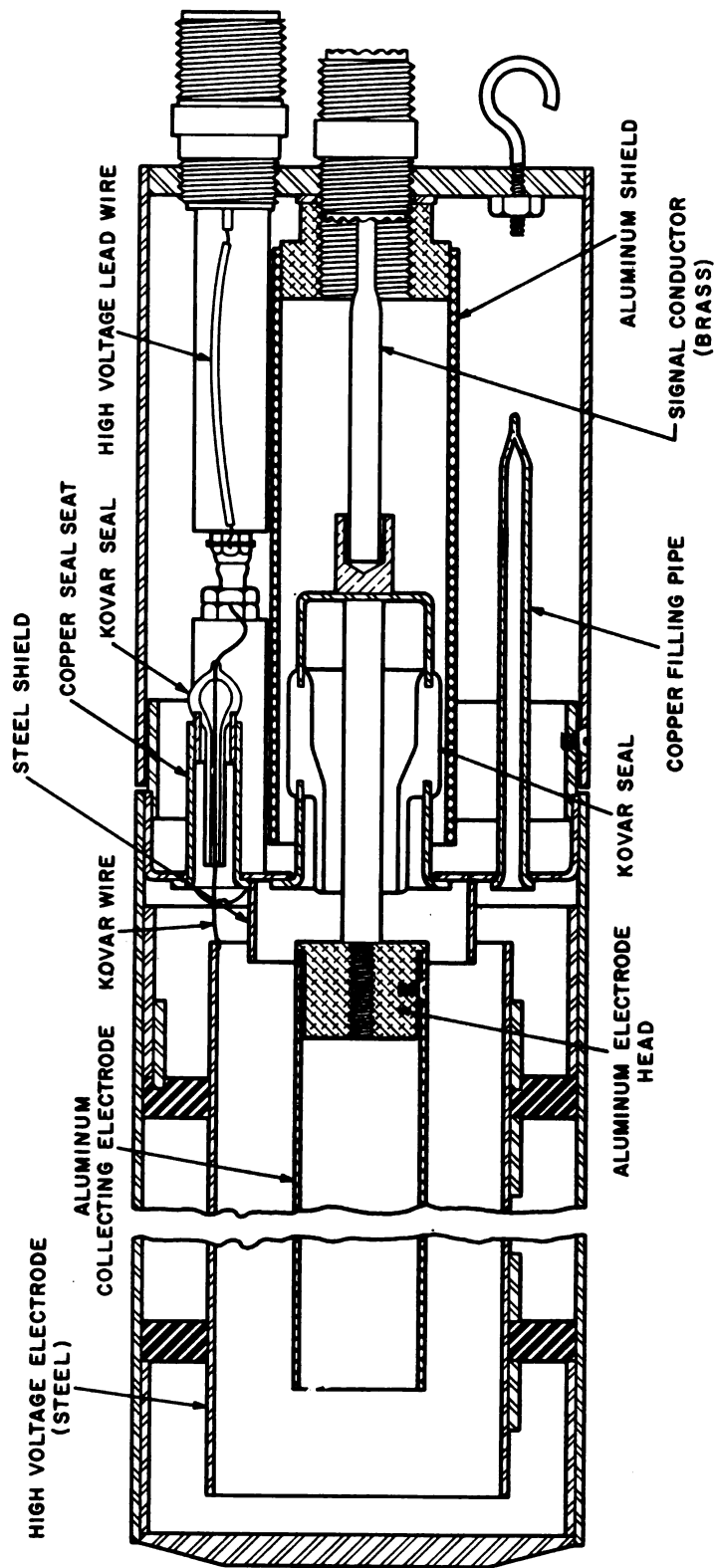


Fig. 6-53 Boron-coated ionization chamber.

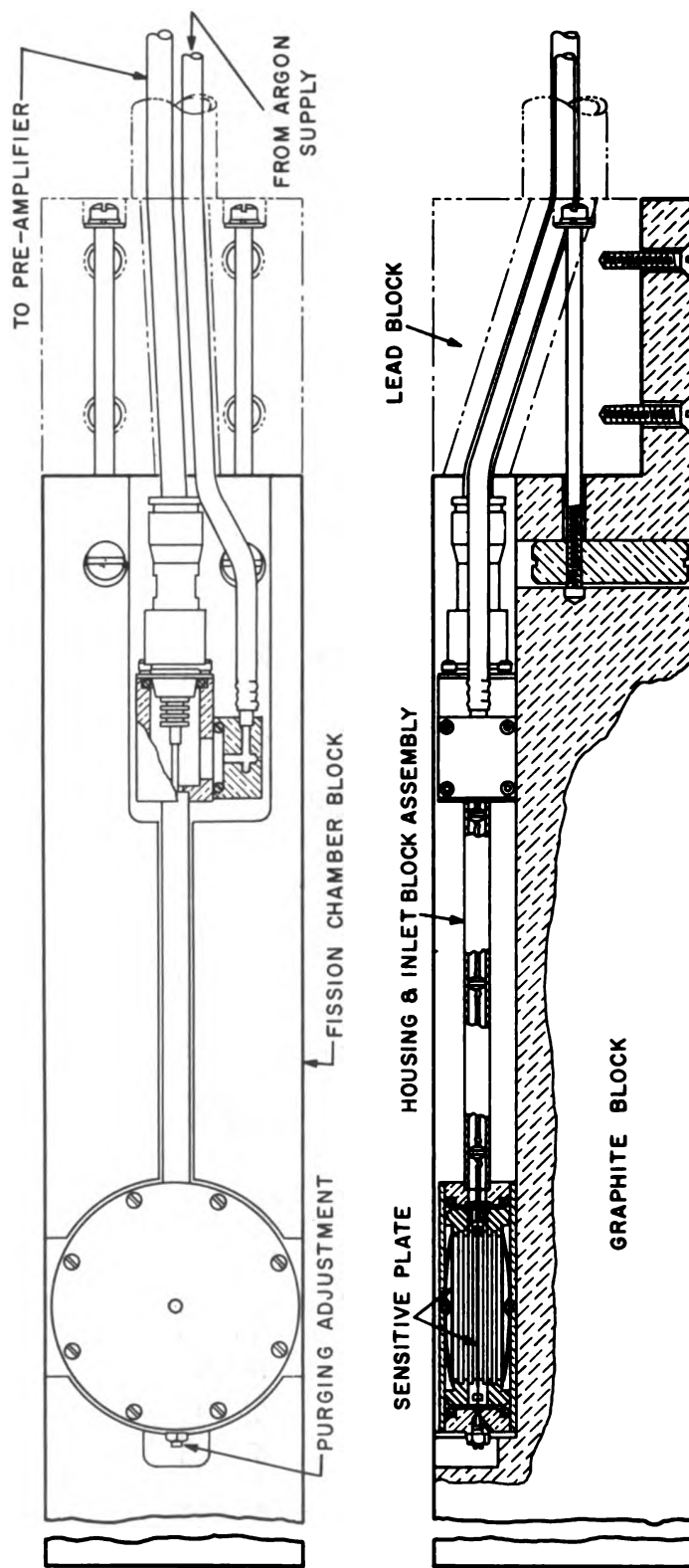


Fig. 6-54 Fission chamber and mounting block.

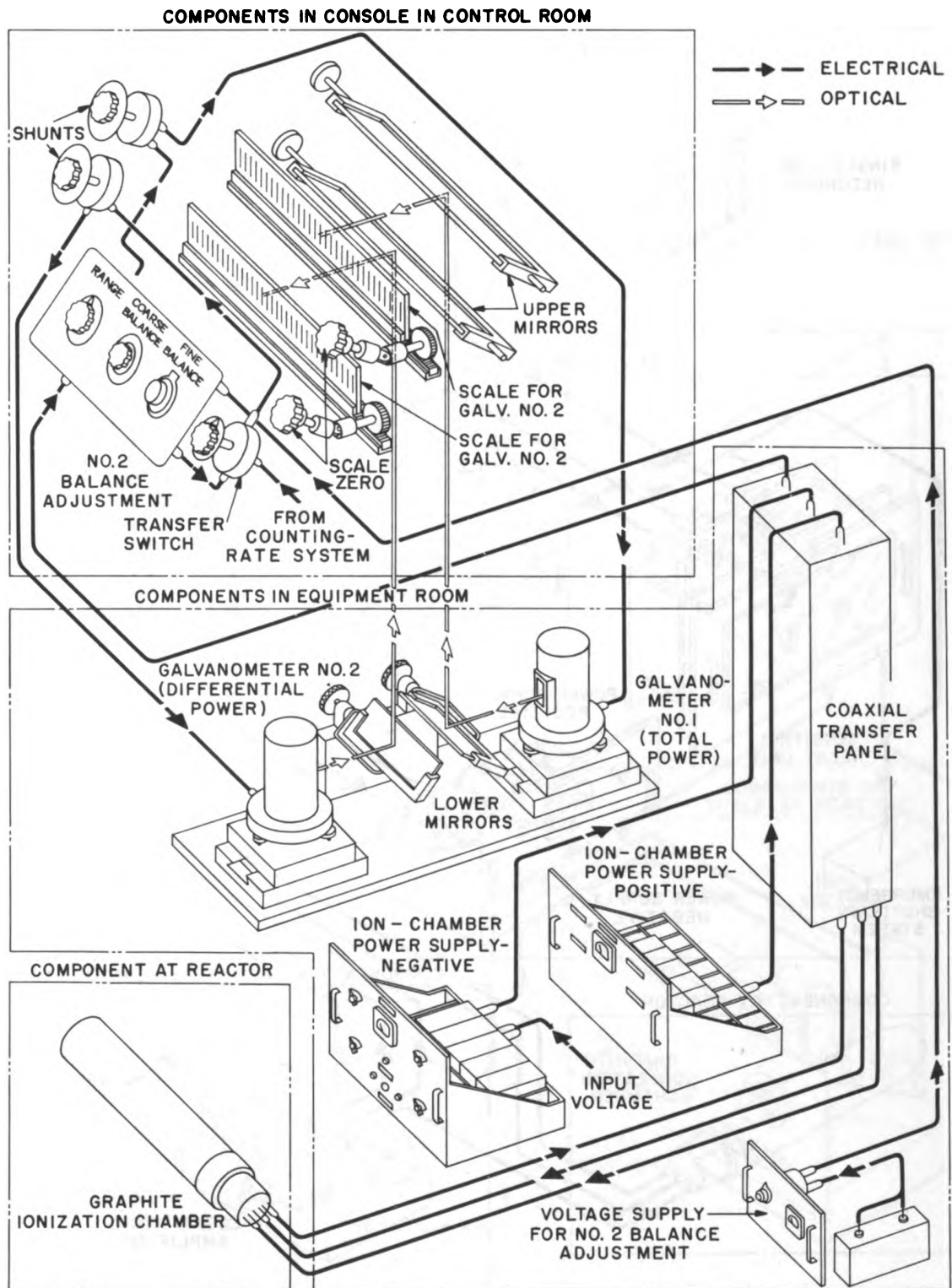


Fig. 6-55 Galvanometer system for indicating reactor power.

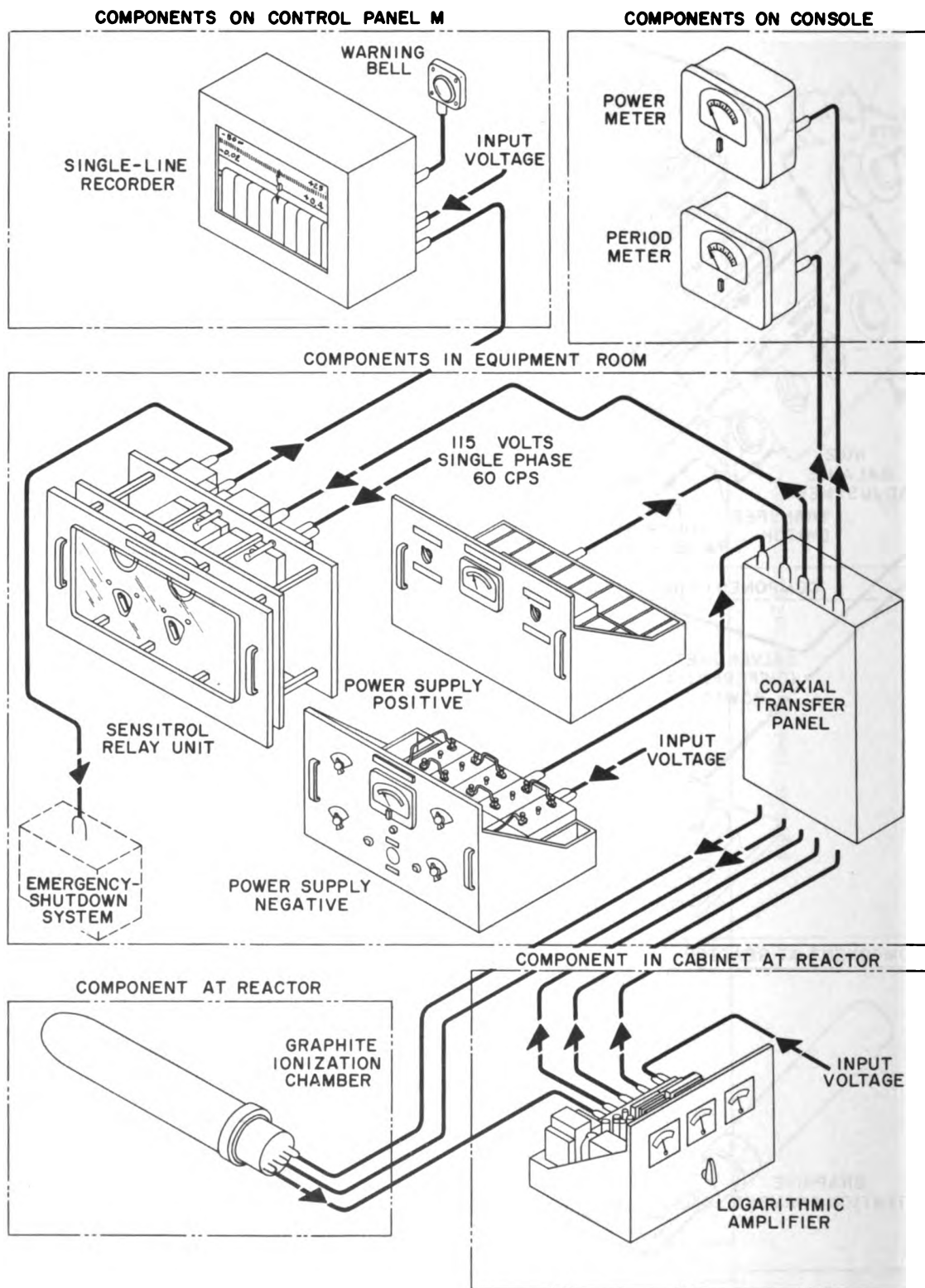


Fig. 6-56 Logarithmic system for indicating reactor period and power.

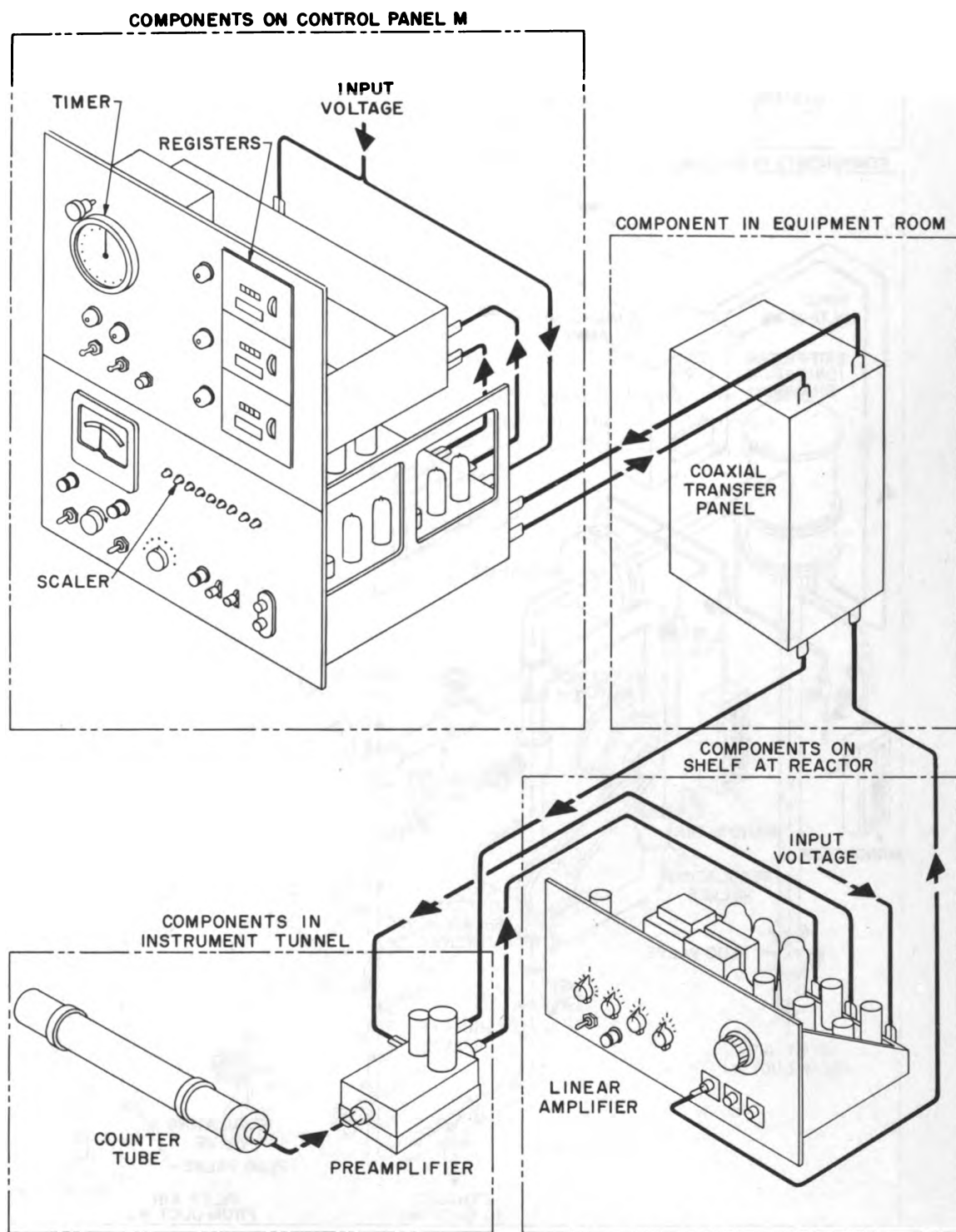


Fig. 6-57 Neutron-counting system.

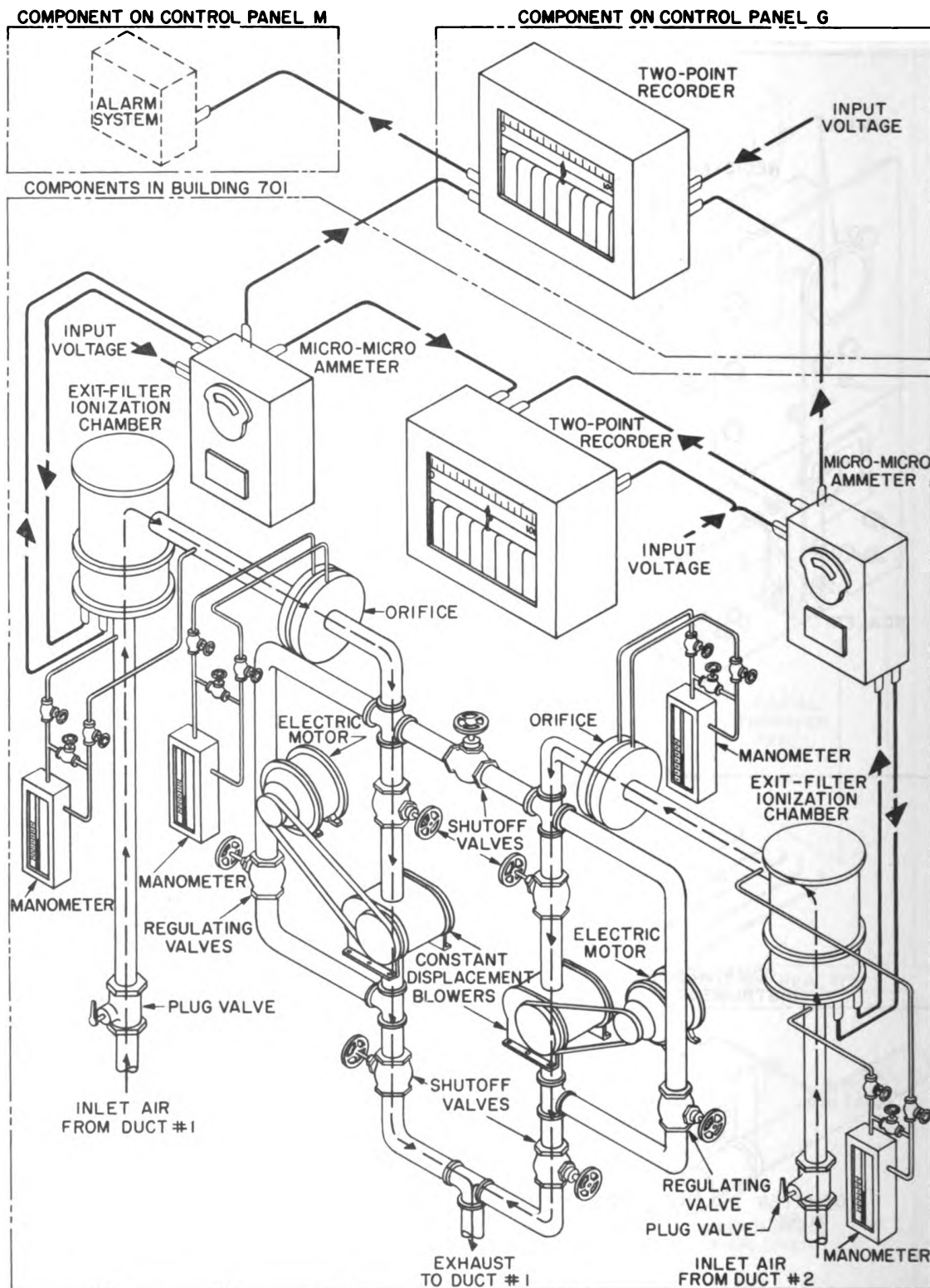


Fig. 6-58 Exit-filter activity system.

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